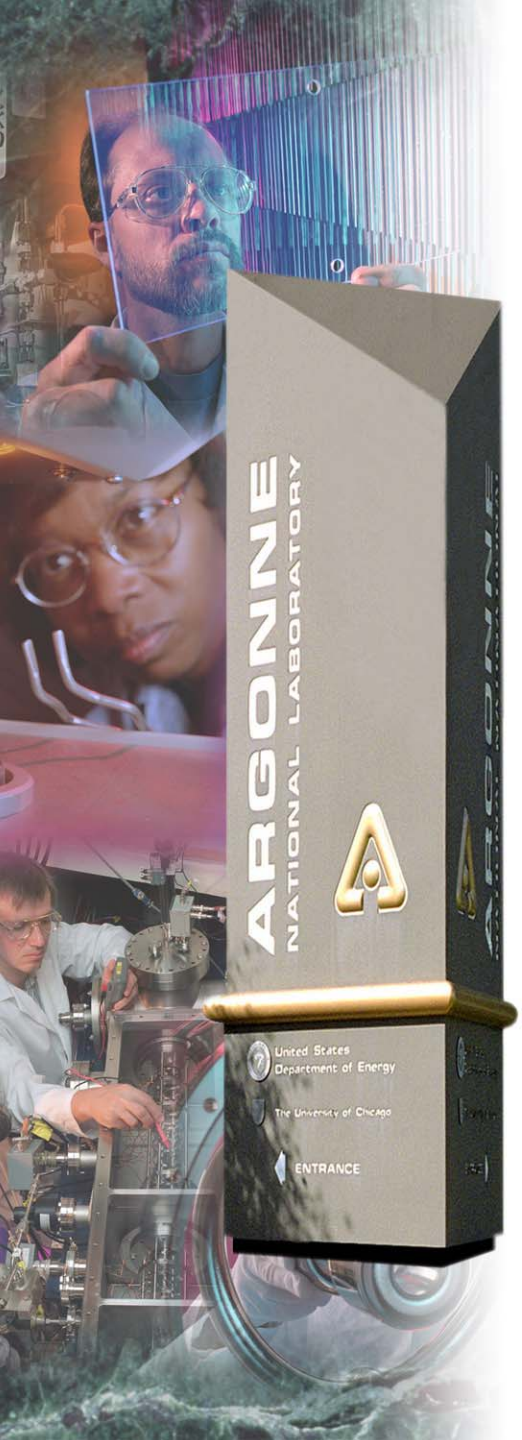


# LOCA Results for Advanced-Alloy and High-Burnup Zircaloy Cladding

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*Energy Technology Division*

**Nuclear Safety Research Conference (NSRC-2003)**
**Washington, DC**
**October 20-22, 2003**

**Argonne National Laboratory**

Office of Science
U.S. Department of Energy

A U.S. Department of Energy
Office of Science Laboratory
Operated by The University of Chicago

# *Scope of LOCA-Relevant Research*

- **Licensing Issues Addressed**
  - 10 CFR 50.46 embrittlement criteria for maintaining residual ductility in Zircaloy (Zry) cladding; temperature limit: PCT ≤ 1204ºC, oxidation limit: effective cladding reacted ECR ≤ 17%
  - Confirm embrittlement criteria for high-burnup Zry-2 and Zry-4
  - Compare post-quench ductility of ZIRLO and M5 to Zry-4 vs. ECR
- **High-Burnup Phenomena Investigated**
  - Fuel behavior and effects of fuel on cladding during a LOCA sequence
  - Effects of corrosion, hydriding and irradiation on cladding: Ballooning, burst, high-temperature steam oxidation, Secondary hydriding, quench behavior and post-quench ductility
- **Advanced-Alloy Cladding Phenomena Investigated**
  - ZIRLO and M5 oxidation kinetics (vs. Zry-4)
  - ZIRLO and M5 post-quench ductility (vs. Zry-4)

# Cladding Alloys and Irradiated Fuel Rods at ANL

- **Unirradiated Cladding Alloys**
  - Zry-2: Zr-lined 8x8, 9x9 (Limerick BWR "archive"); 10x10
  - Zry-4: 15x15 (H.B. Robinson "archive"); 17x17 low-Sn
  - ZIRLO: 17x17
  - M5: 17x17
  - E110: tubing and cladding (etched/anodized or lightly oxidized)
- **High-Burnup Fuel Rod Segments**
  - H.B. Robinson 15×15 PWR rods at 67 GWd/MTU
    Corrosion layer ≤ 110 µm; H-content ≤ 800 wppm
  - Limerick 9×9 BWR rods at 56 GWd/MTU
    Corrosion layer ≈ 10 µm; H-content ≈ 70 wppm

# *Advanced-Alloy Post-Quench Ductility Research*

- **Basic Approach**
  - **Short (25-mm), undeformed cladding segments**

    Oxidize (2-sided)-and-quench all alloys in same apparatus

    Use measured weight gain to determine oxygen pickup and ECR

    Perform RT ring-compression tests to determine ductility

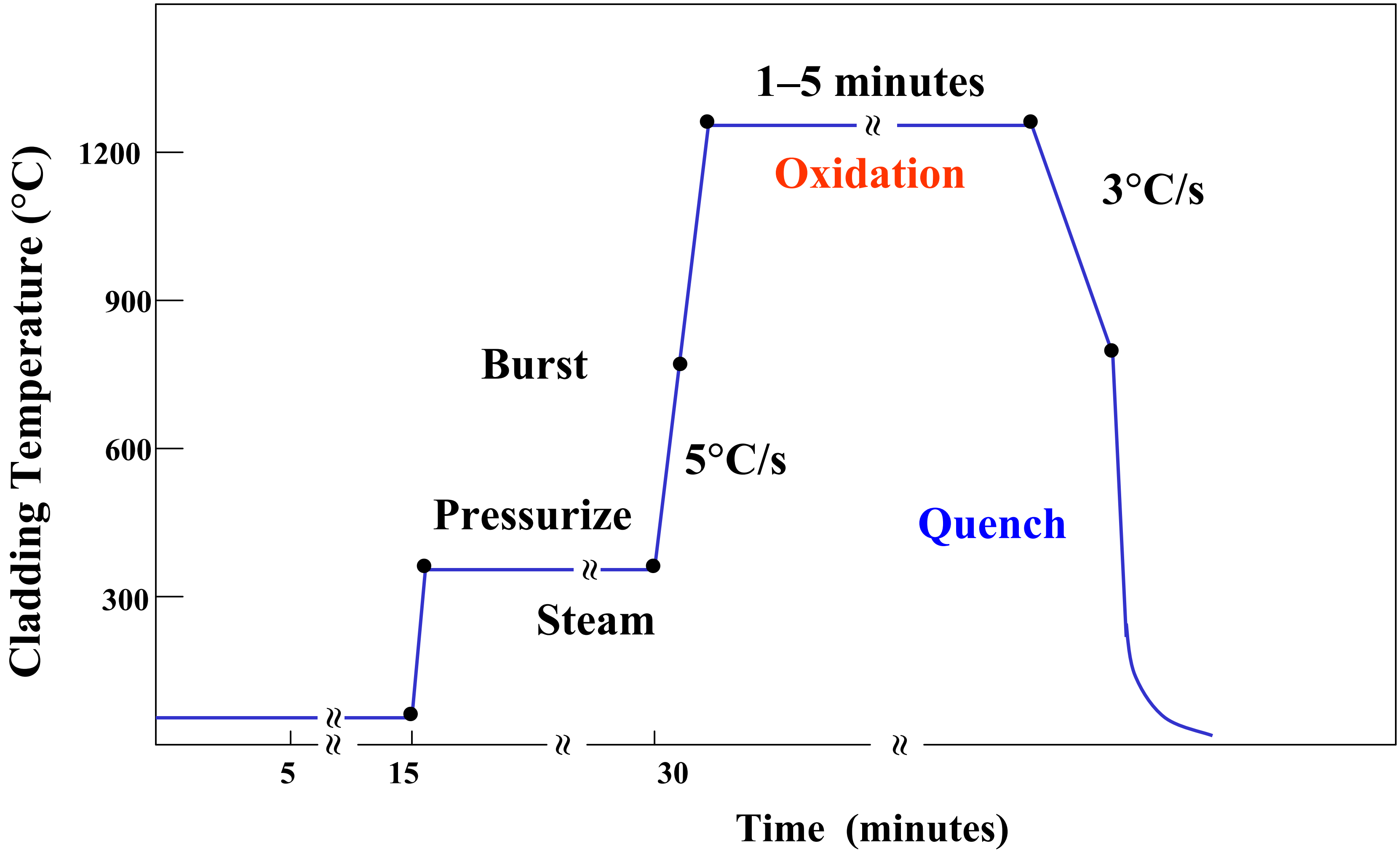
    Use metallography and LECO H-determination to confirm results
  - **Long (300-mm), pressurized cladding segments**

    Balloon, burst, oxidize and quench all alloys in same apparatus

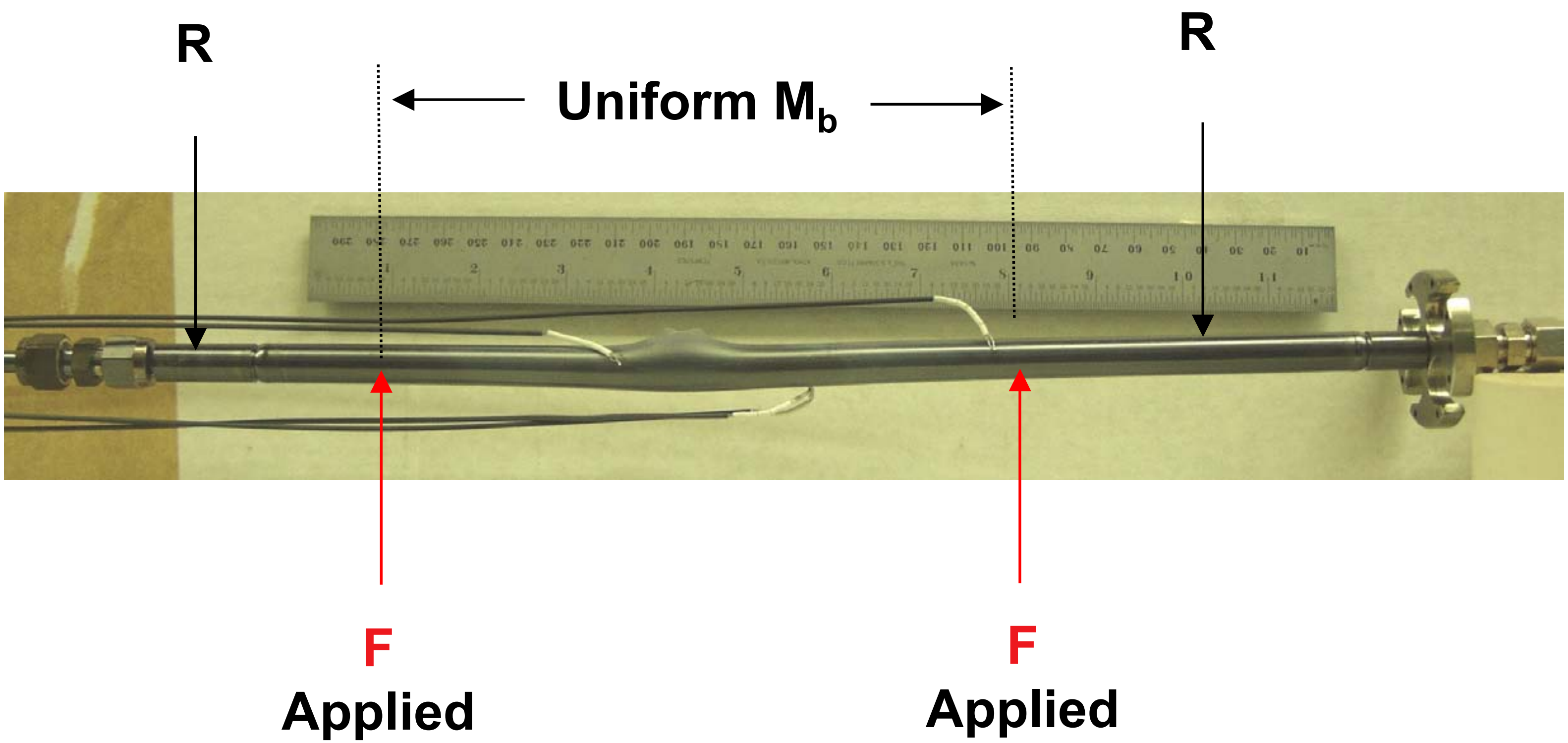
    Use 4-point-bend test to determine failure location and mode

    Perform ring-compression tests on samples from non-ballooned region
- **Oxidation Times and Temperatures for Calc. ECR ≤ 20%**
  - ≤3400 s (1000°C), ≤1100 s (1100°C), ≤400 s (1200°C), ≤230 s (1260°C)

# LOCA Integral Test Sequence for Unirradiated Cladding Alloys



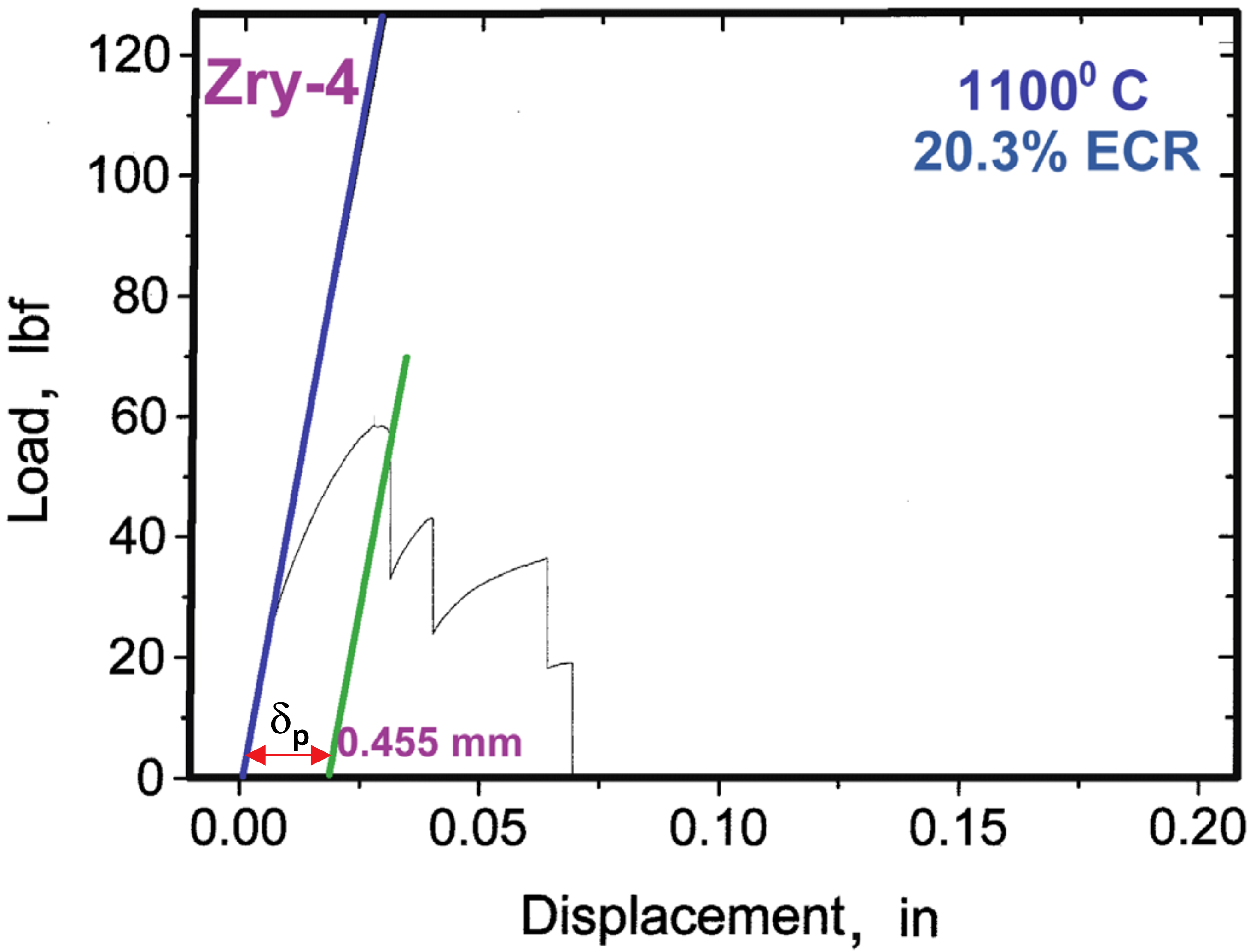
# 4-Point-Bend Test with Burst Area under Tension



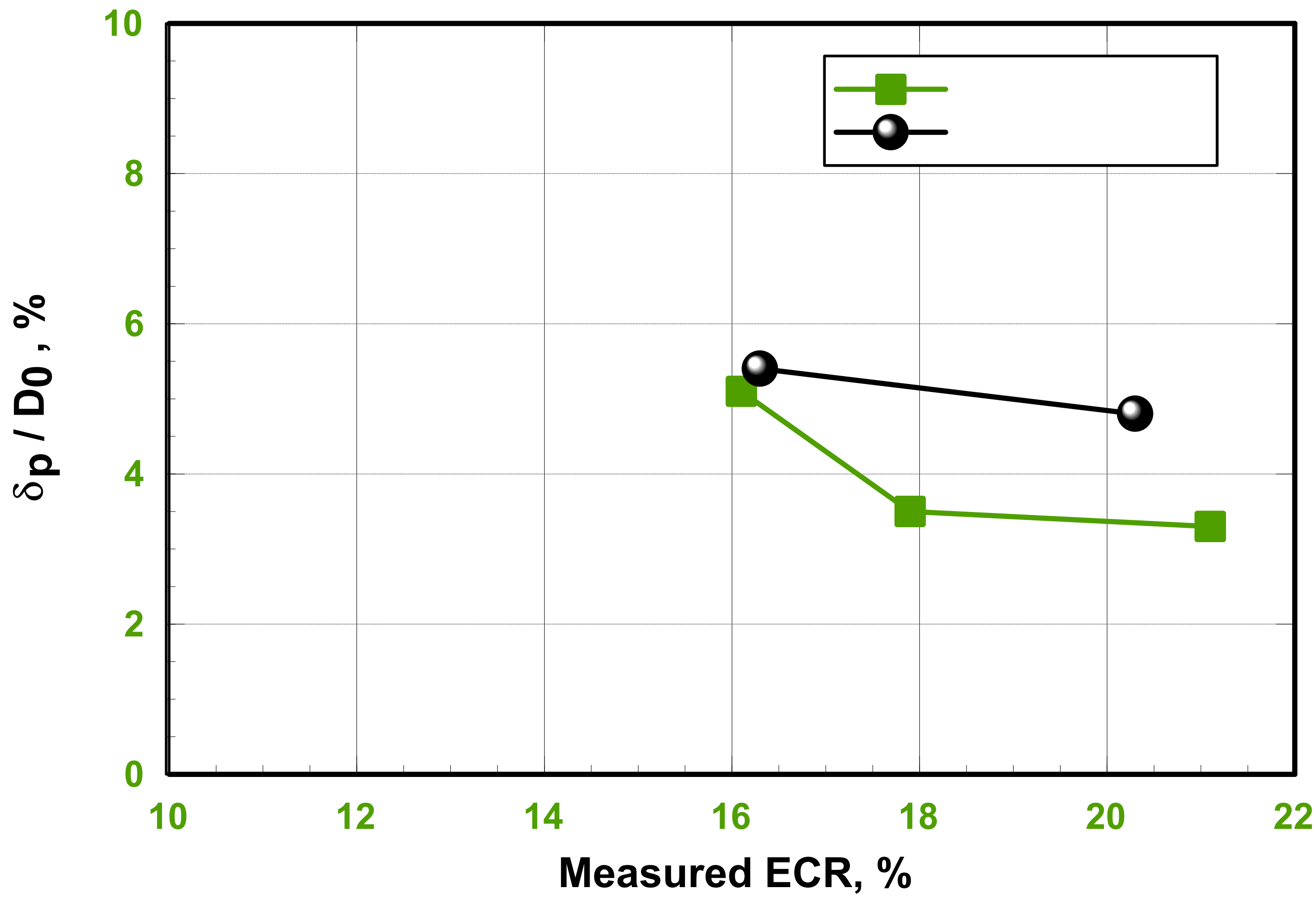
# *Advanced-Alloy Results*

- **Weight Gain Kinetics for Short Segments**
  - At 1100ºC, Zry-4, M5 and ZIRLO data are in agreement with Cathcart-Pawel (CP) model predictions (within ≈ ±10%)
  - At 1000ºC, ZIRLO < Zry-4 and M5 << Zry-4
- **Post-Quench Ductility for Short Segments**
  - Residual ductility measured for all alloys oxidized at 1000ºC and 1100ºC
  - Measured hydrogen pickup is low (< ≈100 wppm)
  - Metallography conducted to date supports ductility data
- **Tests at 1200°C and 1260°C are in Progress**
- **LOCA Integral Tests on Pressurized, Long Segments**
  - Conducted on 9×9 Zry-2 at 1204ºC for 5 minutes (18-20% ECR)
  - Highly non-uniform local "ECR" observed at burst cross-section
  - Significant secondary hydriding observed in balloon neck regions

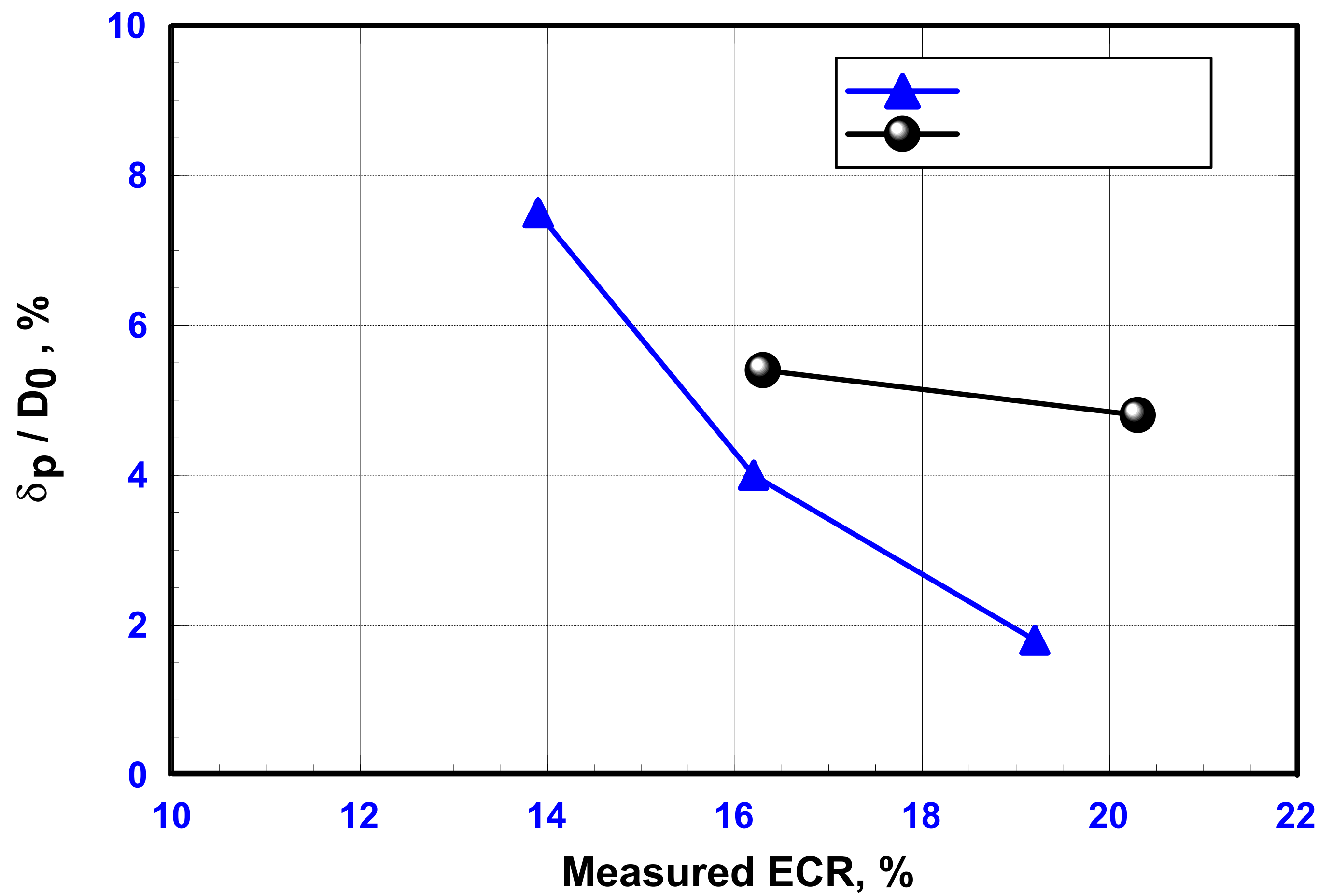
# RT Load-Displacement: Zry-4 after 20% ECR at 1100°C



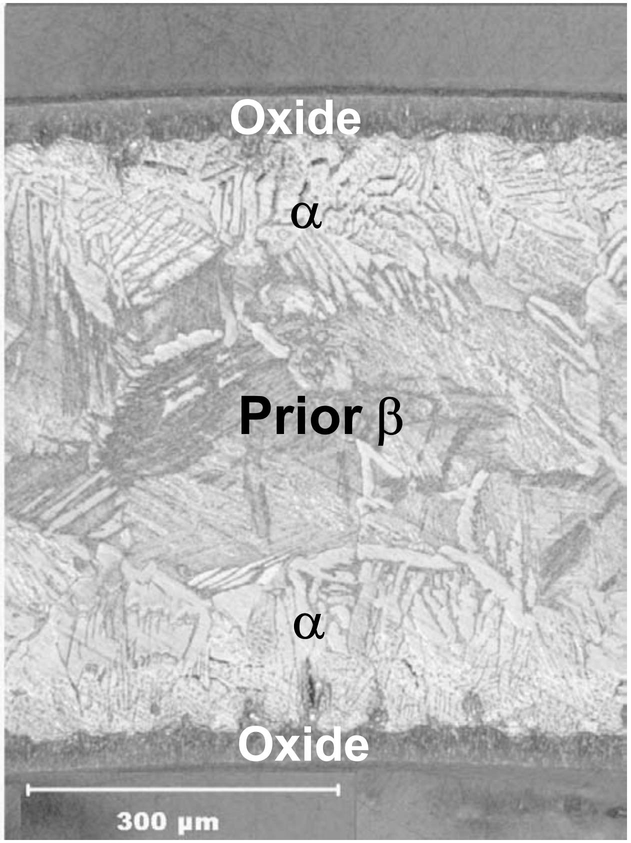
# RT Ring-Compression Ductility of ZIRLO & Zry-4 Samples Oxidized at 1100°C and Quenched at 800°C

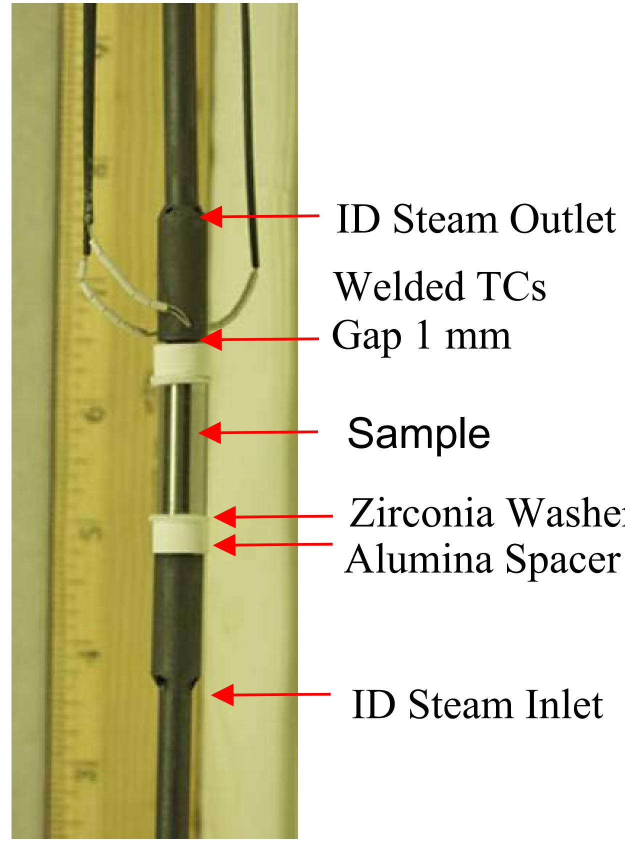


# RT Ring-Compression Ductility of M5 & Zry-4 Samples Oxidized at 1100°C and Quenched at 800°C

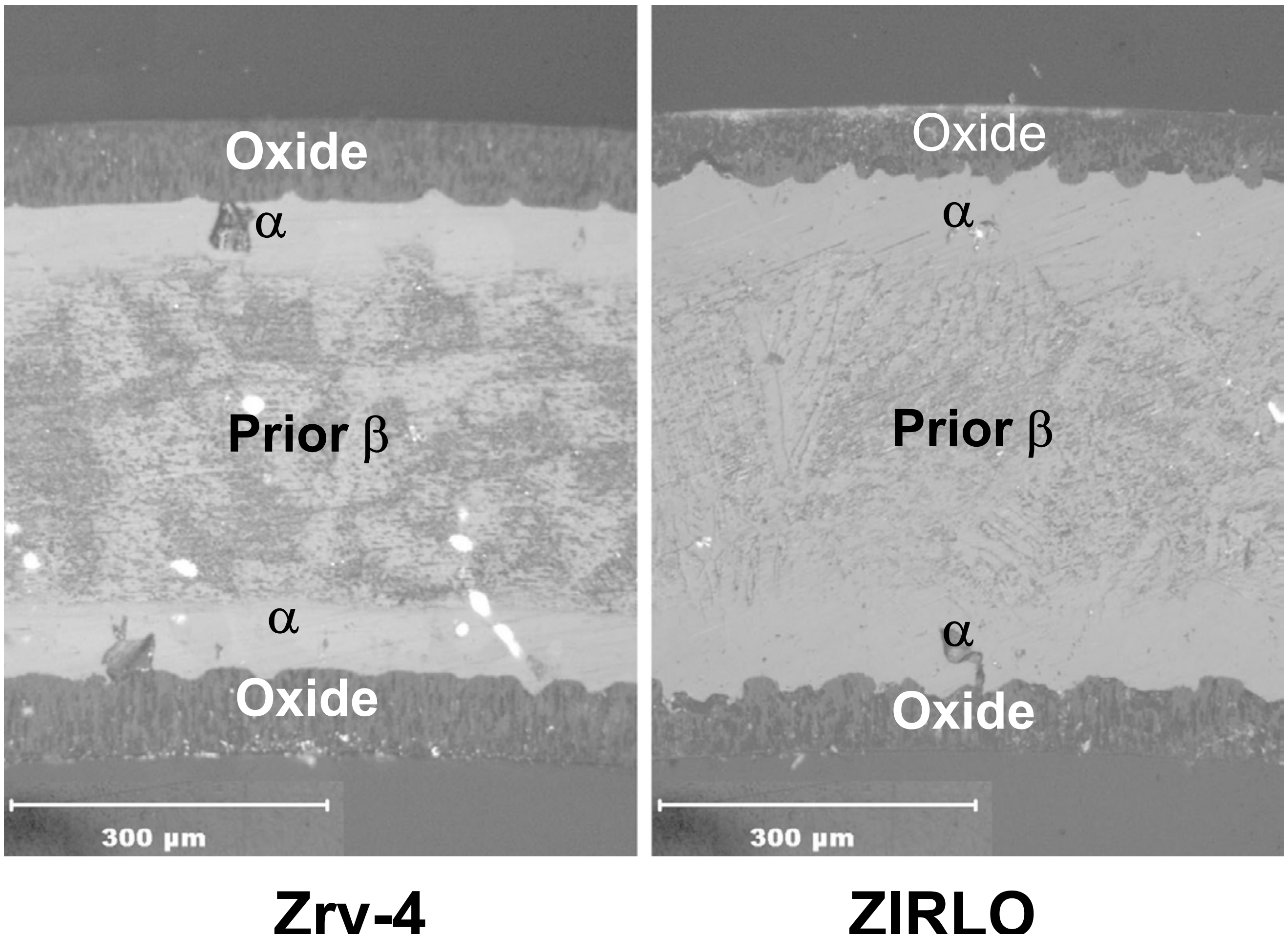


# *Metallography of M5 Oxidized at 1000ºC to CP-Model-Calculated ECR = 20% (13% Measured ECR)*

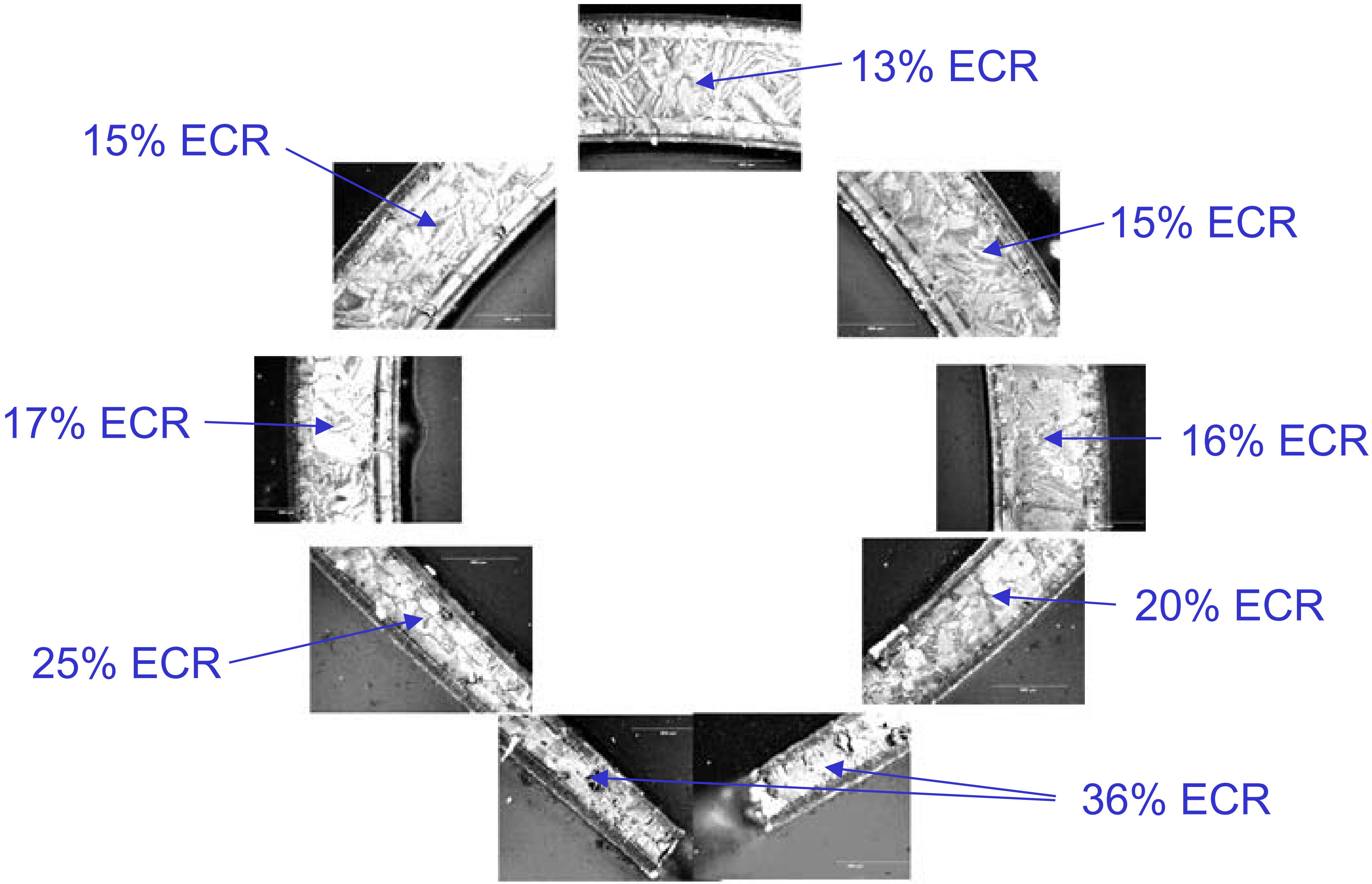


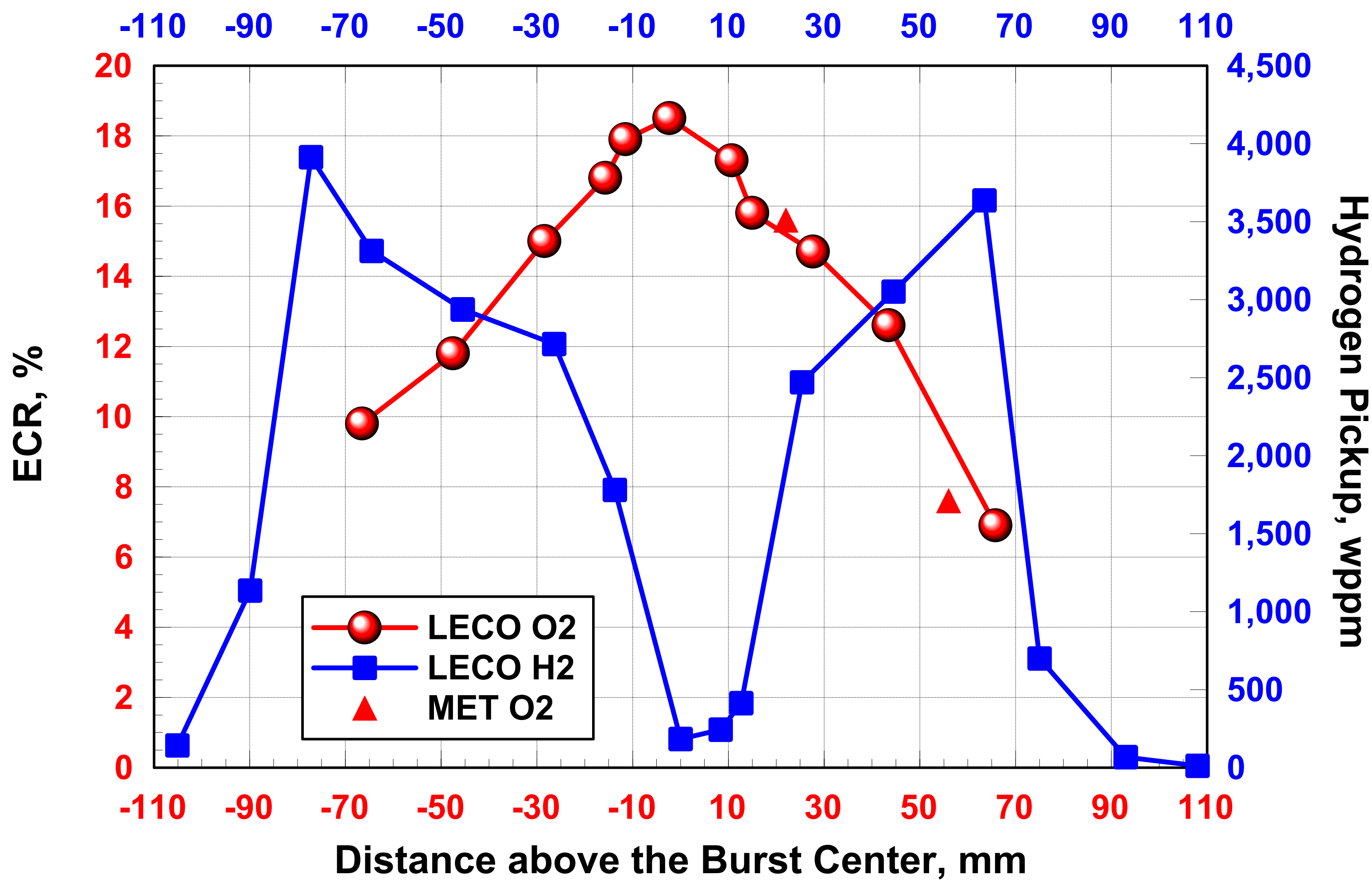
# *Metallography of Zry-4 and ZIRLO Oxidized at 1000ºC to CP-Model-Calculated ECR = 20%*



# OCL#11 Burst Cross-Section: 1204°C, 5 min., 18% ECR



# *LOCA Integral Test Results for Zry-2: 1200°C for 5 Min.*

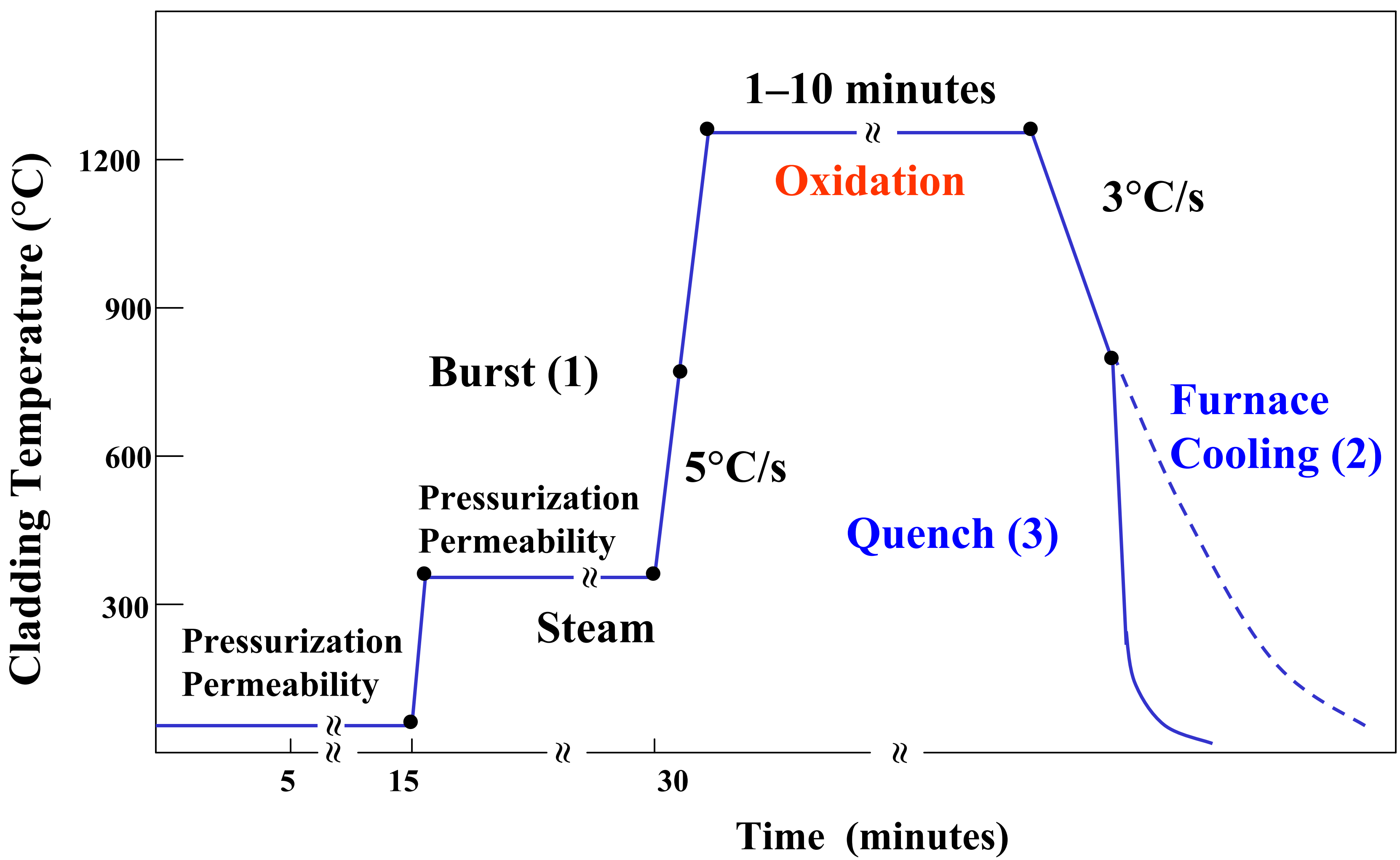


# *LOCA Integral Test Results for Limerick Zry-2*

- **Temperature History**
  - Stabilize at 300°C and 1200 psig (8.3 MPa) internal pressure
  - Ramp at 5°C/s through ballooning & burst to 1204°C
  - Hold for 1-10 minutes, cool to 800°C at 3°C/s and quench
- **Detailed Examinations**
  - **Profilometry, metallography, H & O determination (in progress)**
  - 4-Point-Bend Tests & Ring-Compression Tests (to be conducted)
- **Post-Quench-Ductility Demonstration Tests with Unirradiated Zry-2 Oxidized to 18-30% ECR at 1204°C**
  - Brittle failure of 10-min. sample (30% measured ECR) in burst region at 100°C following quench due to dead weight loading
  - 4-point-bend-test & handling failures for 18-20% ECR specimens

    Brittle failure observed in burst (O-embrittled), balloon (O- and H-embrittled) and neck (H-embrittled) regions

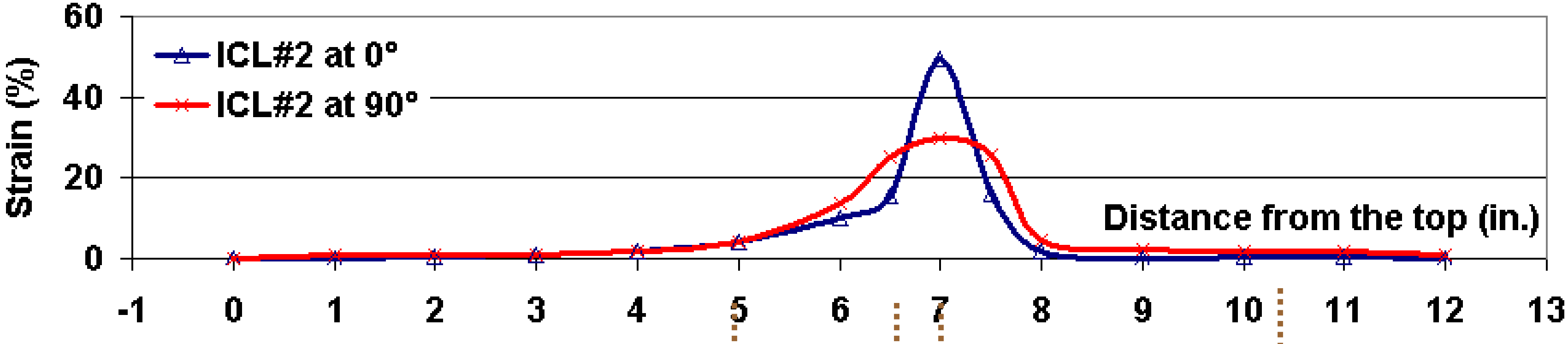
# *LOCA Integral Test Sequence*

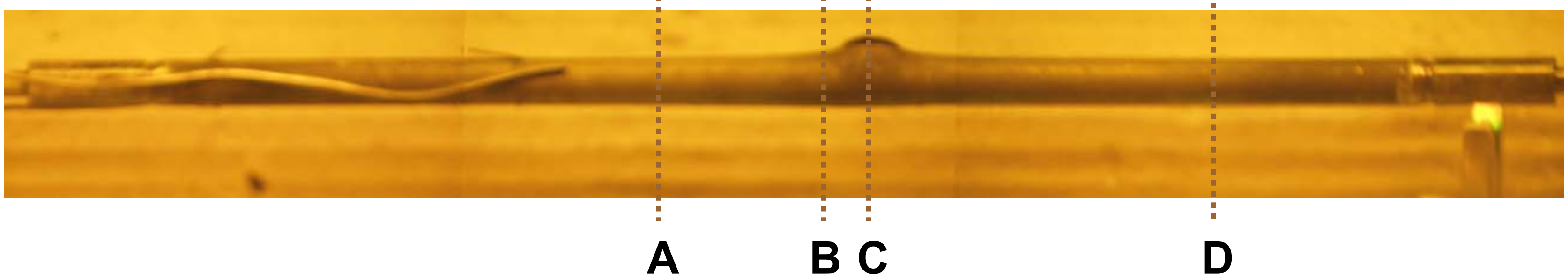


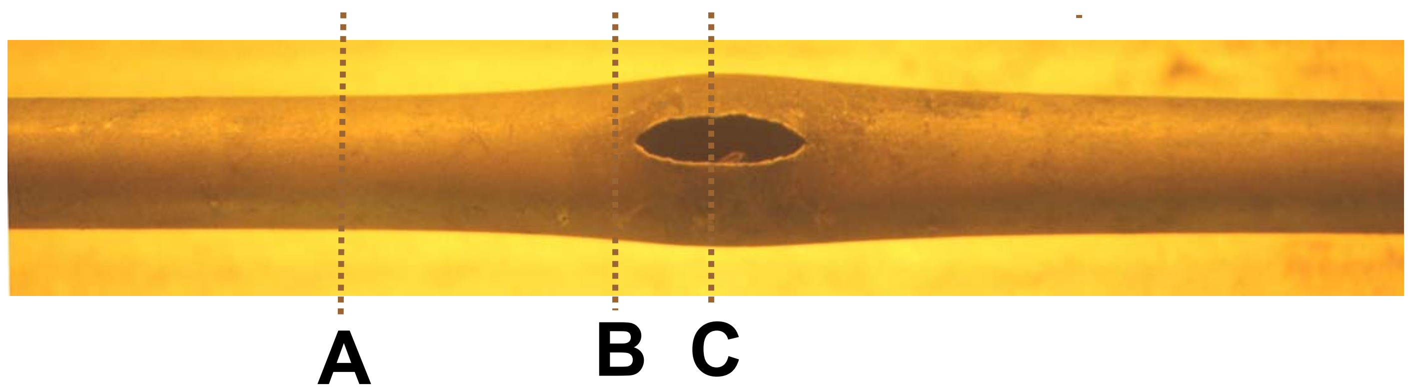
# *LOCA Integral Tests*
# *High-Burnup BWR Fuel Specimens*

- **Ramp-to-Burst Test Conducted in Argon (ICL#1)**
  - T- and P-histories, photos and profilometry reported at NSRC-2002
  - With exception of burst shape (oval) and axial extent of ballooning (shorter), results are similar to those for unirradiated Zry-2
- **LOCA Sequence with 5-minute Oxidation at 1204°C and Slow-Furnace Cooling (ICL#2)**
  - T- and P-histories, photos and profilometry reported at NSRC-2002
  - With exception of burst shape (oval) and axial extent of ballooning (shorter), results are similar to those for unirradiated Zry-2
  - **Additional fuel and cladding characterization has been performed**
- **LOCA Sequence with Quench (ICL#3, Nov. 2003)**
  - Specimens have been prepared
  - Quench system is being added to in-cell apparatus

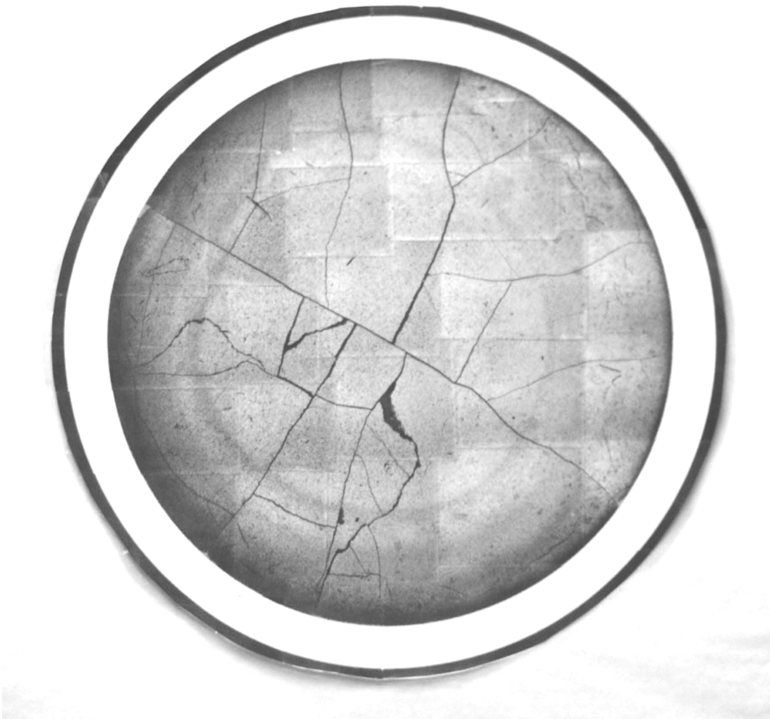
# *Locations of Metallographic Samples for ICL#2 Specimen*



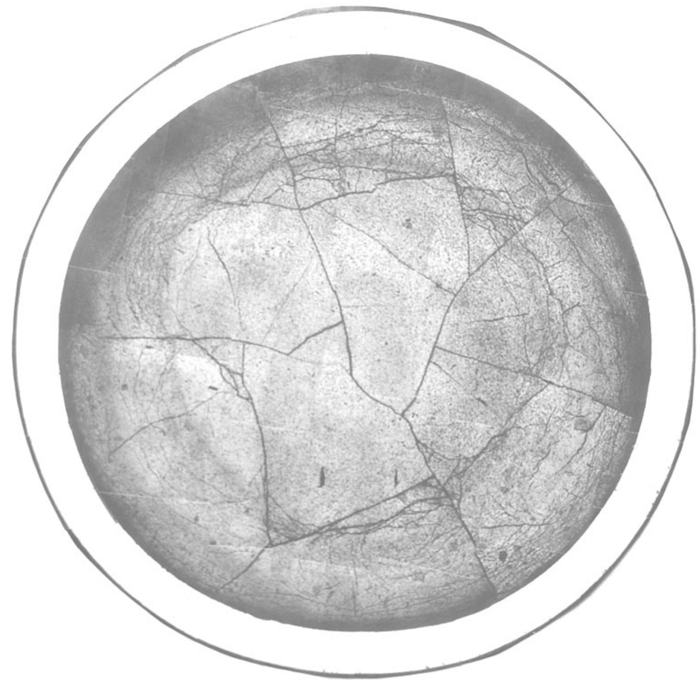



# Fuel Metallographic Results for ICL#2 Specimen

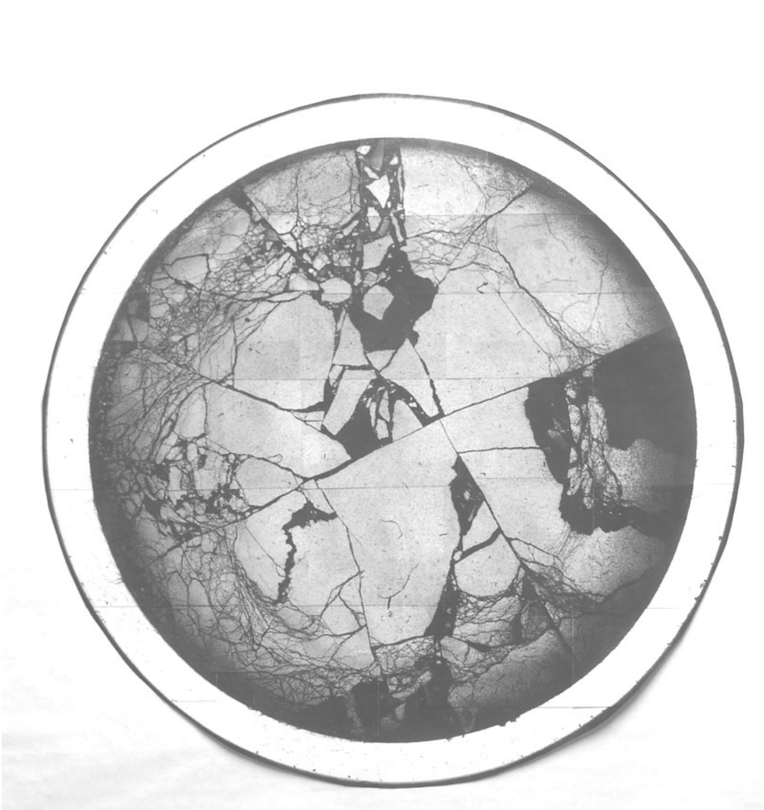


As-received Specimen:
180 mm away from the LOCA sample

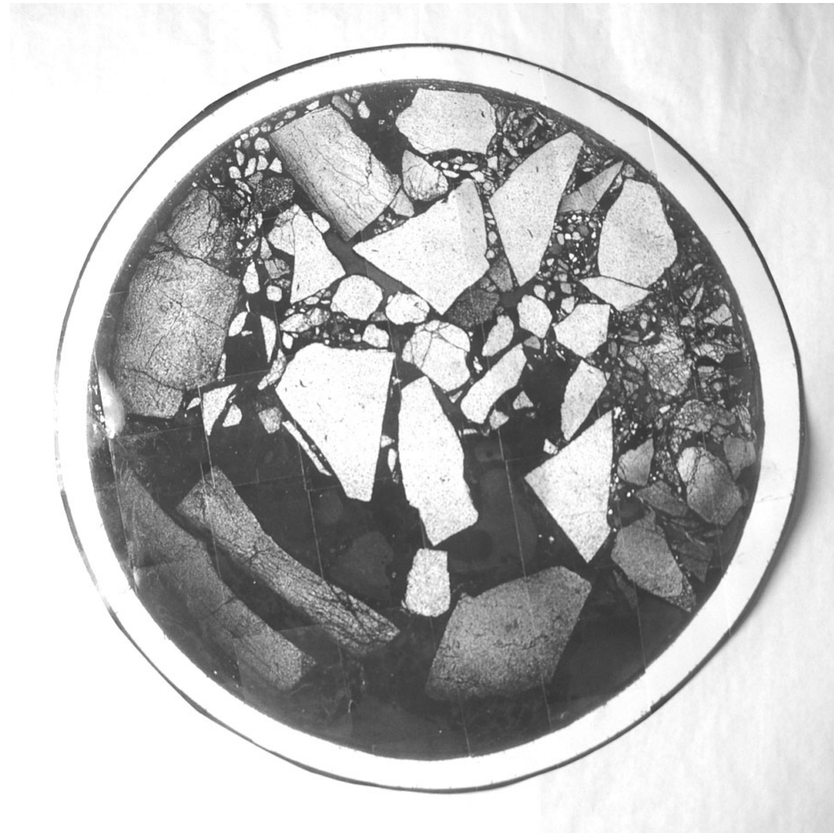


Post-test Specimen D:
≈45 mm to the bottom end-cap
Strain: 0 - 2%

# Cladding Metallographic Results for ICL#2 Specimen

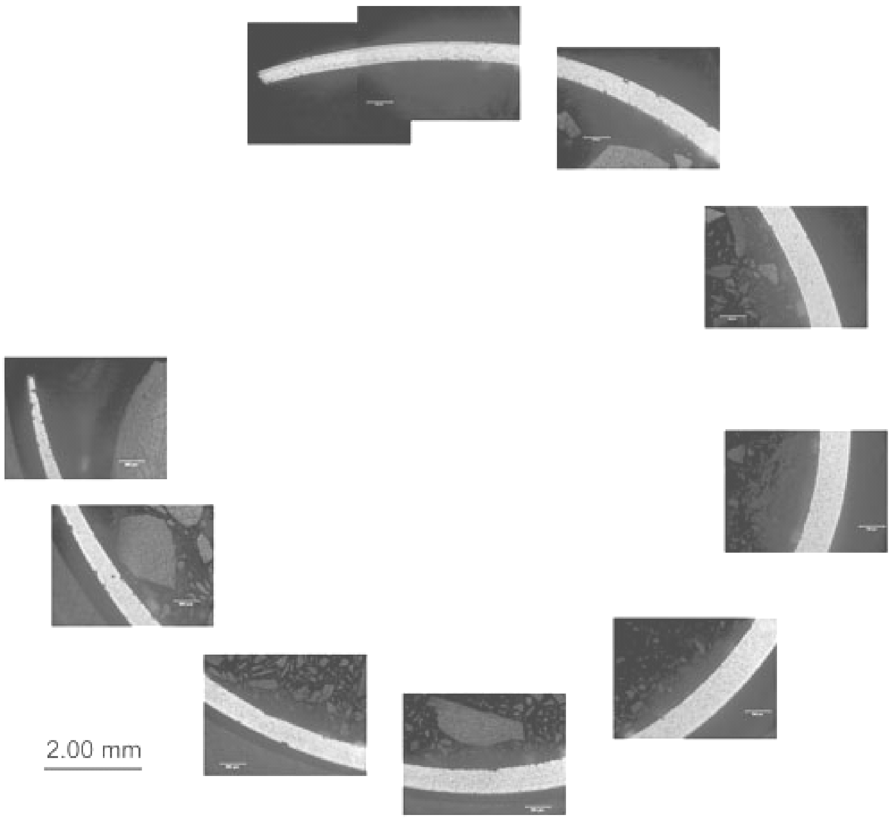


Post-test Specimen A:
≈ 50 mm above the burst mid-plane
Strain: 2% - 4%

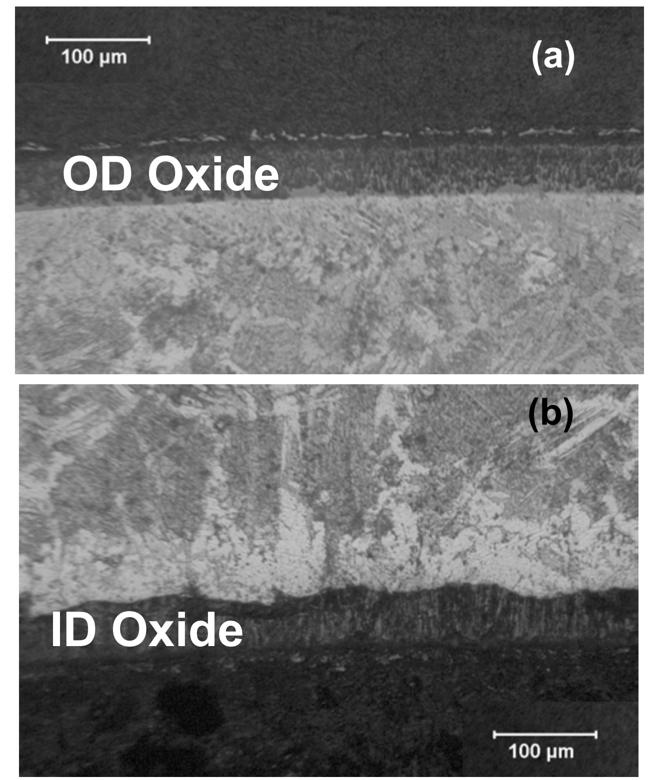


Post-test Specimen B:
≈12 mm above the burst mid-plane
Strain: 15% - 25%

# *Burst Cross-Section for High-Burnup ICL#2 Test*

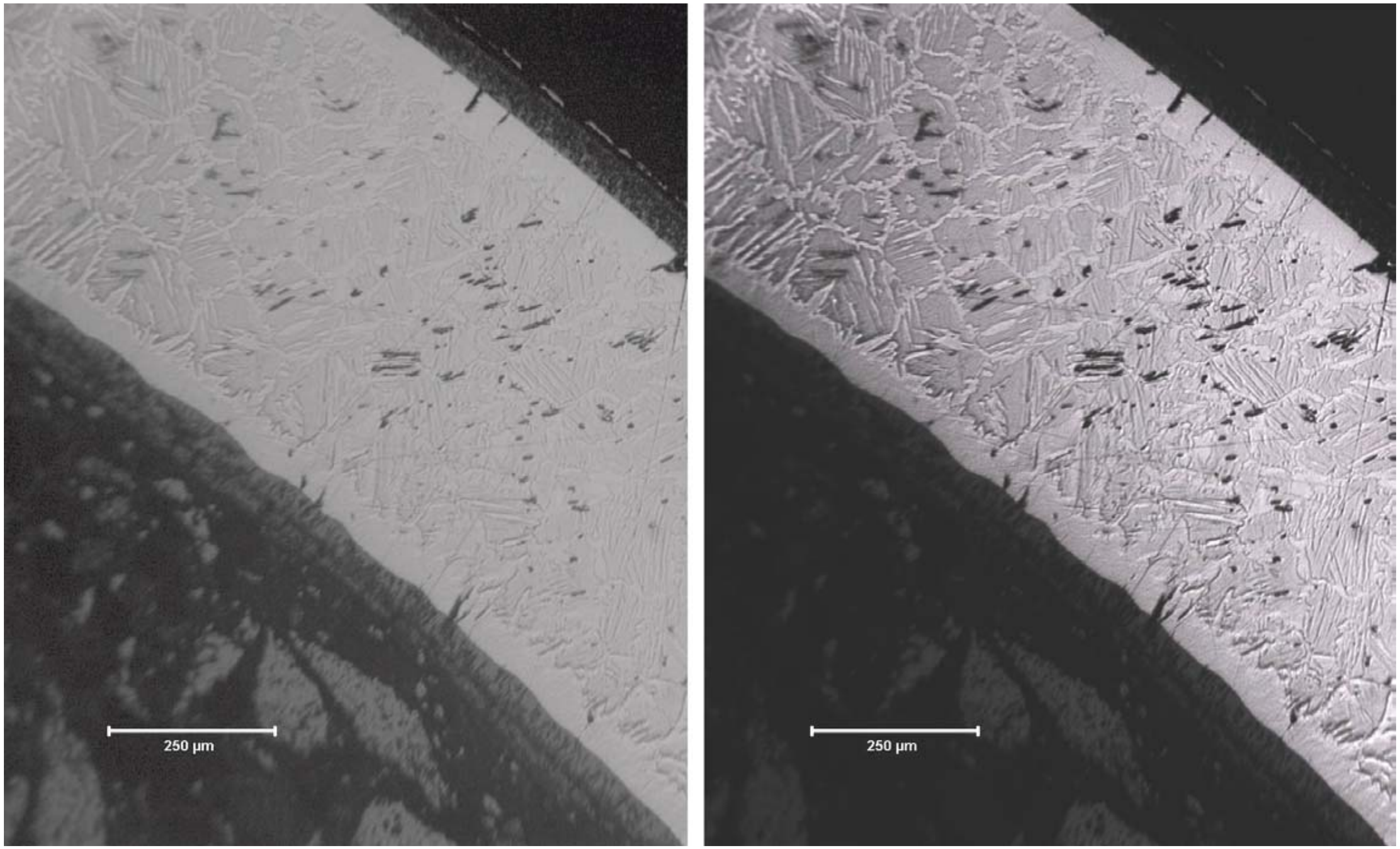



Post-test  Specimen C:
Burst mid-plane
Strain: 30% - 50%

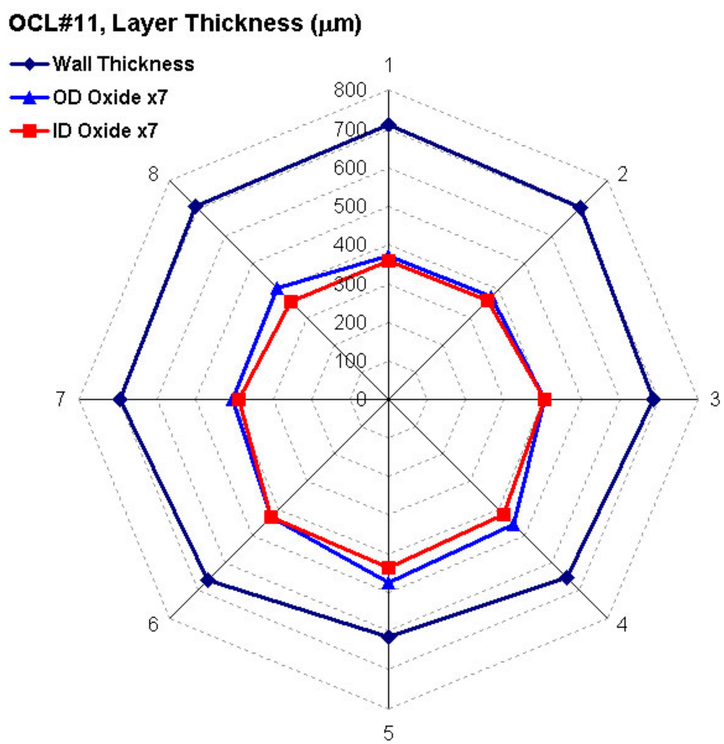


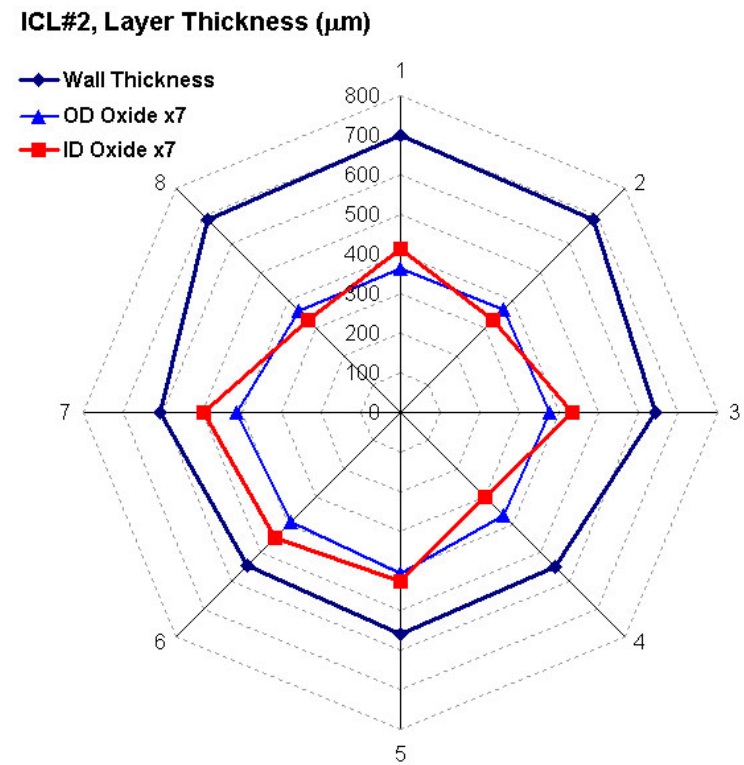

High magnification images of (a) outer and (b) inner surfaces of Specimen C.

# ICL#2 (1204°C, 5 Min.) Cladding Metallography
# 12 mm above Burst Center



# Variation in Oxide Layer Thickness: OCL#11 vs. ICL#2 12 mm above the Burst Center




Axial Location: ≈ 18 mm above the burst
Ave. Wall Thickness: ≈638 (μm)
Weight Gain (MET.): 23 (mg/cm$^2$)

Axial Location: ≈ 12 mm above the burst
Ave. Wall Thickness: ≈ 622 (μm)
Ave. Weight Gain (MET.): 21 (mg/cm$^2$)

# *Future LOCA-Relevant Work*

- **Advanced-Alloy Post-Quench Ductility**
  - Oxidize-and-quench 1200ºC and 1260ºC samples (Zry-4, ZIRLO, M5)
  - Conduct ring-compression tests; H measurements & met for 20% ECR
  - Conduct LOCA Integral Tests with advanced-alloy cladding samples
- **In-Cell LOCA Integral Tests with High-Burnup Samples**
  - Conduct Limerick BWR tests (3-5 min. at 1204ºC) with quench
  - Initiate Robinson PWR oxidation and LOCA tests
  - Develop simple in-cell 4-point-bend test benchmarked to out-of-cell Instron tests; perform bend test on fueled post-quench samples
  - Perform ring compression tests on defueled samples from beyond the ballooned region; use tabletop Instron in beta-gamma cell
- **Continue Companion Out-of-cell LOCA Integral Tests**

# LOCA Behavior of E110 Alloy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***PRESENTED BY L. YEGOROVA***


**Nuclear Safety Research Conference**
**Washington, D.C., USA**
**October 20–22, 2003**

# Outline of the Program First Stage

1. Status of the work

During 2001–2002 Nuclear Safety Institute of Russian Research Center "Kurchatov Institute" (NSI RRC KI) in cooperation with Russian State "Research Institute of Atomic Reactors" (RIAR) with the support of Joint Stock Company "TVEL" (Russian Federation), U.S. Nuclear Regulatory Commission (USA), and Institute for Radiological Protection and Nuclear Safety (IRSN, France) carried out the first stage of the program to reassess the post-quench ductility of Russian zirconium-niobium alloys under the LOCA conditions

2. Program purpose

- to determine the embrittlement threshold of the Russian type of Zr-1%Nb cladding (E110 alloy) as a function of the ECR
- to determine the sensitivity of the E110 embrittlement threshold to:
  - oxidation scenario (temperature, heating and cooling rates, one-sided and two-sided oxidation types);
  - alloying components (oxygen (E110K alloy), iron, tin (E635 alloy))

3. Major provisions of the program and analysis of test results

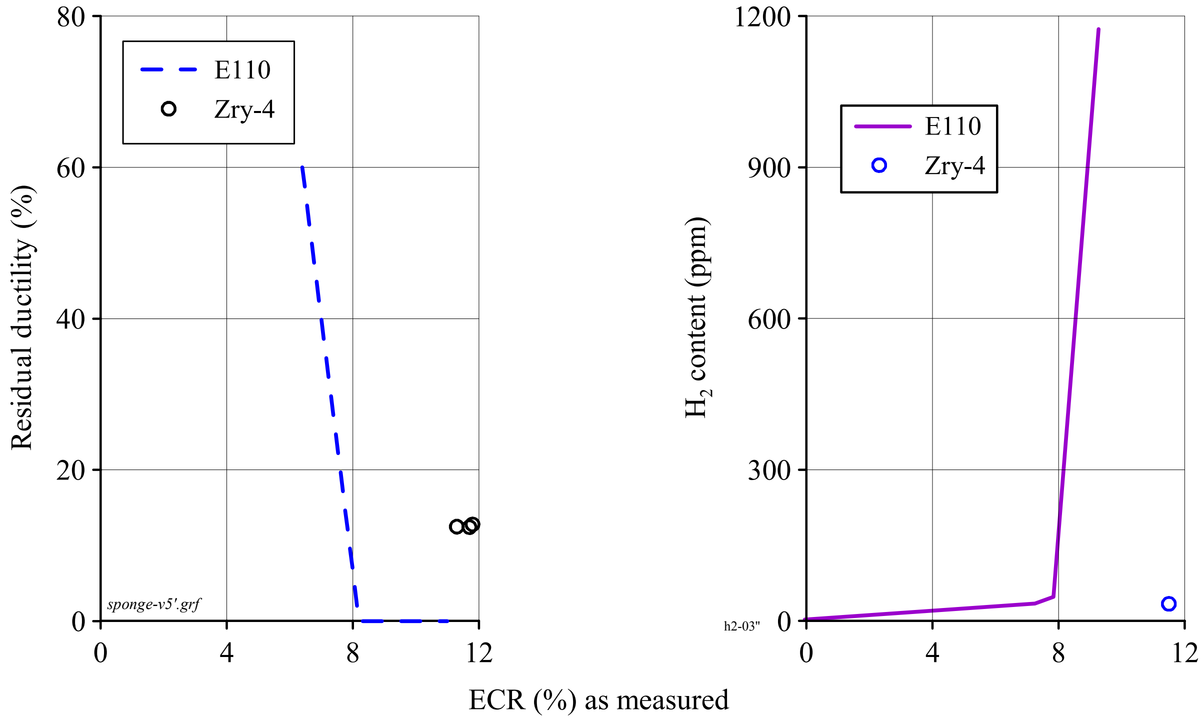
See: V.Asmolov et al., "Understanding LOCA-Related Ductility in E110 Cladding", Proceedings of the 2002 Nuclear Safety Research Conference, NUREG/CP-0180, March 2003

# Background of the Work at the End of the Program First Stage

**Comparative data to characterize residual ductility and hydrogen concentration as a function of the ECR for E110 and Zry-4 alloys**

↓

## Double-sided oxidation at 1100 C




↓ ↓

**8% ECR as measured corresponds to 11% ECR as calculated if Russian E110 conservative kinetics (Bochvar institute correlation) or Baker-Just Zry-4 conservative correlation is used**

↓

### Major conclusions:

- the zero ductility threshold of the E110 alloy is lower than that of Zry-4 alloy
- an earlier initiation of the breakaway effect in the nodular corrosion form and intensive absorption of hydrogen as a direct consequence are the visible reasons for the different behavior of these two alloys
- the correlation between the ductility threshold and such alloying elements as O, Fe, Sn was not observed

# Discussion of the Program First Stage

1. Subject of the discussion

- M5 (France) and E110 (Russia) claddings are manufactured from the similar alloys on the basis of the Zr-1%Nb composition
- the comparison of published French data on the oxidation behavior and embrittlement threshold of the M5 cladding with the appropriate results of this study with E110 cladding allows to reveal the following general differences:
  - the embrittlement thresholds of M5 and Zry-4 alloys are similar
  - the embrittlement threshold of E110 alloy is lower than that for the M5 and Zry-4 alloys
  - unlike that of the E110 cladding, the embrittlement of the M5 cladding is not accompanied by the nodular corrosion and hydrogen uptake

2. Possible explanations of results

First version: listed differences were caused by differences in experimental procedures and approaches used to interpret test results

Second version: the similar alloying composition of zirconium-niobium alloys does not assure their identical oxidation and mechanical behavior under LOCA relevant conditions

# Discussion of the Program First Stage

3. <u>Decisions</u>

1. To obtain the comparative test data in the same apparatus by the same technicians on the basis of ANL program with M5, Zirlo, E110, Zry-4 alloys
2. To develop coordinated ANL / RRC KI program to reveal the response of the E110 cladding to the variation of the manufacture processes

# Characterization of the Factors Selected for Studies at the Program Second Stage

| Type of possible factors | Specification | Approaches to demonstrate the sensitivity of test results to different factors | Involved laboratories | |
|---|---|---|---|---|
| | | | RRC KI/RIAR | ANL |
| 1. Surface effects | Surface roughness and surface contamination | To polish, machine, etch the cladding surface | + | + |
| 2. Bulk effects | Chemical composition of Zr ingot | To use the sponge Zr ingot and Zr ingot with low Hf instead of iodide and electrolytic Zr ingots | + | – |
| | Microstructure effects (grain size, phase composition, secondary precipitates (composition, size, distribution) as a function of the fabrication process | To perform comparative SEM, TEM examinations for different types of cladding specimens | + | + |
| 3. Geometrical sizes | Typical cladding thickness of E110 cladding is 0.69–0.71 mm. Typical cladding thickness of M5 cladding is 0.56–0.6 mm | To machine the E110 cladding to the M5 size | – | + |

# Surface Effects Studies

Types of new cladding specimens:

1. E110 etched and anodized standard Russian cladding
2. E110 polished (inner and outer surface) standard Russian as-received tubing

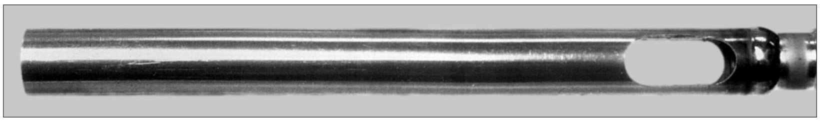
**Demonstration of the oxidation behavior of etched and anodized cladding**

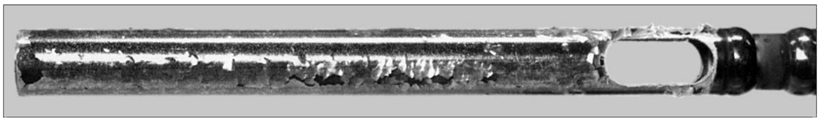
↓

**1100 C, double-sided oxidation, F/F**

↓

**Appearance of the cladding before (a) and after (b) the oxidation test**

a) 

b) 

↓

**Conclusion:**
The current final chemical processing of the E110 cladding does not improve the cladding oxidation behavior

↓

**Additional comments:**

- This conclusion fully agrees with results of tests with the etched E110 cladding performed by ANL
- Russian vendor of VVER fuel (JSC "TVEL") eliminates the final etching of the VVER cladding from the standard conditioning procedures

# Surface Effects Studies

**Demonstration of the oxidation behavior of etched and anodized cladding**

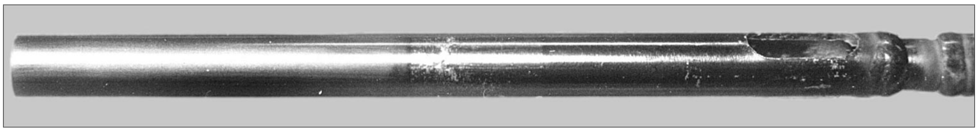
↓

**1100 C, double-sided oxidation, F/F**

↓

**Appearance of the cladding after the oxidation test**

polished ↓ as-received ↓



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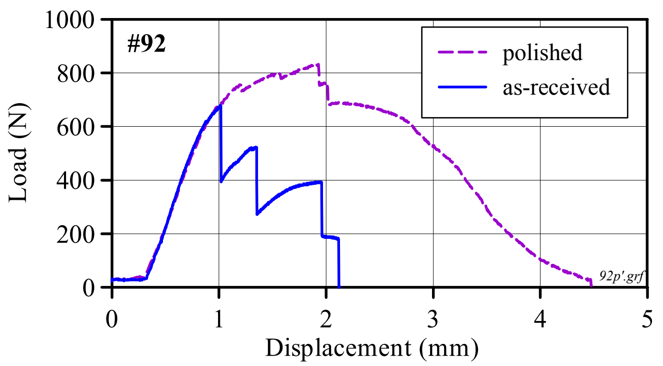
**Results of ring compression tests**

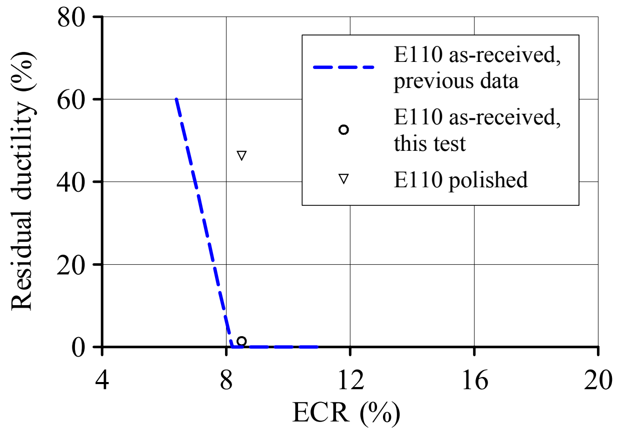



**Conclusion:**

**Polishing of the cladding tubes leads to:**

- significant delay in the start of nodular oxidation
- sufficient improvement of the ductility

↓




**Comments:**

**ANL independent data confirms this conclusion**

# Bulk Effect Studies. Introduction

1. General statement:

Let's assume that M5 (Zr-1%Nb) alloy is really better than E110 (Zr-1%Nb) alloy under LOCA relevant conditions

2. General question based on this statement

What is the reason for different oxidation behavior of these two alloys?

3. Summary of possible answers and comments

| Variant of answers | Comments |
|---|---|
| 1. Different oxygen concentrations in these alloys (0.04–0.06% in E110, 0.13% in M5) lead to the different oxidation behavior | Special studies of E110K alloy (oxygen concentration 0.13%) have demonstrated that the increase of oxygen concentration does not change the oxidation behavior of E110 alloy |
| 2. Different conditioning of cladding surface (polishing for M5, etching for E110) leads to the different oxidation behavior | Surface effect studies of E110 polished specimens have shown that the appropriate manipulation with cladding surface allows to slow down the initiation of the nodular corrosion, but this procedure does not allow to avoid this type of the oxidation in the given range of ECRs (especially at 1000 C) |

# Bulk Effect Studies. Introduction (continued)

| Variant of answers | Comments |
|---|---|
| 3. Differences in the chemical composition of Zr ingot | 1. All types of alloys developed for PWR reactors (including advanced zirconium-niobium alloys) are manufactured on the basis of the sponge Zr. Therefore, the differences in the chemical composition of these family of alloys are conditioned by differences in the alloying composition<br>2. Cladding alloys for the VVER reactor are manufactured on the basis of the mixture of iodide and electrolytic Zr<br>3. Thus, the chemical composition of the PWR cladding and VVER cladding with the same alloying components differ in the composition of impurities<br>4. To verify the sensitivity of E110 oxidation behavior to the chemical composition of impurities, it was decided to perform the oxidation and mechanical tests with advanced types of E110 claddings manufactured on the basis of the modified process of Zr ingot fabrication |
| 4. Differences in the cladding fabrication process | 1. It is known that cladding properties (mechanical and oxidation parameters) are a strong function of the fabrication process<br>2. Besides, it is known that a strong correlation is observed between the cladding microstructure and cladding oxidation behavior<br>3. Thus, if the differences in the oxidation behavior of zirconium-niobium alloys are conditioned by the cladding fabrication process, the comparison of appropriate microstructures will allow to reveal this effect |

# Bulk Effect Studies. Cladding Types

Types of E110 cladding material used for the bulk effect studies

| Alloying composition | Components of Zr ingot | Conventional name (in the frame of this work) |
|---|---|---|
| 1. Zr-1%Nb | iodide Zr, electrolytic Zr, recycled scrap | E110 |
| 2. Zr-1%Nb | French sponge Zr | G110-f |
| 3. Zr-1%Nb | First step: Russian sponge Zr + iodide Zr + recycled scrap were crushed to powder<br>Second step: This powder was used to manufacture sponge Zr ingot | G110-3ru |
| 4. Zr-1%Nb | French sponge Zr + iodide Zr + recycled scrap | G110-3f |
| 5. Zr-1%Nb | iodide Zr + electrolytic Zr with low Hf, recycled scrap | E110-low Hf |

# Bulk Effect Studies. Zr Ingot Variations

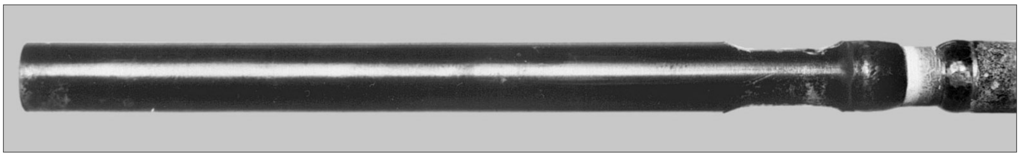
## Test results with the G110-f cladding (E110 alloy on the basis of French sponge zirconium)

**1100 C, double-sided oxidation, F/F, 10.5–13% ECR**

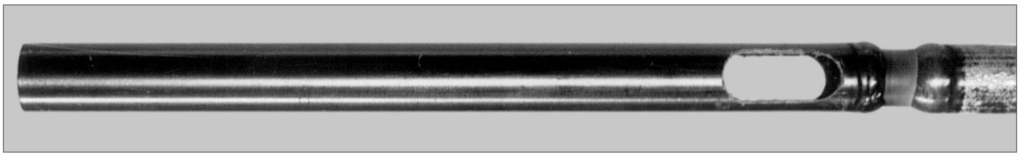
↓

**Appearance of the claddings after the oxidation test**

**Sample #89**
**10.5% ECR**
**($H_2 \rightarrow$22 ppm)**



**Sample #90**
**13% ECR**
**($H_2 \rightarrow$30 ppm)**



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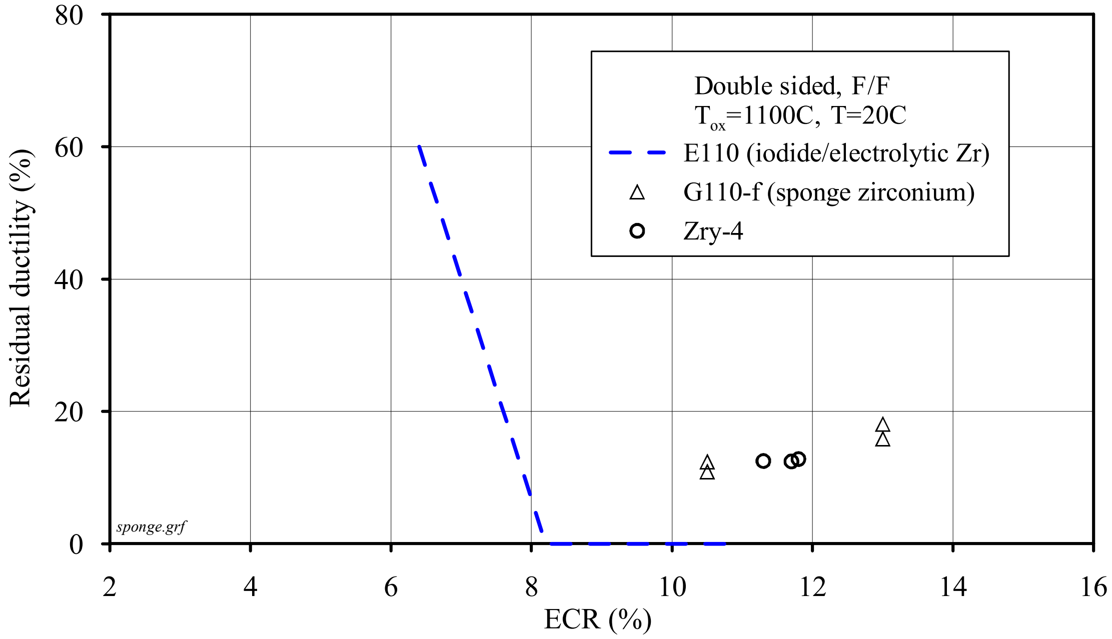
**Comparison of the residual ductility vs. the ECR for E110, G110, and Zry-4 claddings**




↓

**<u>Conclusion:</u>**

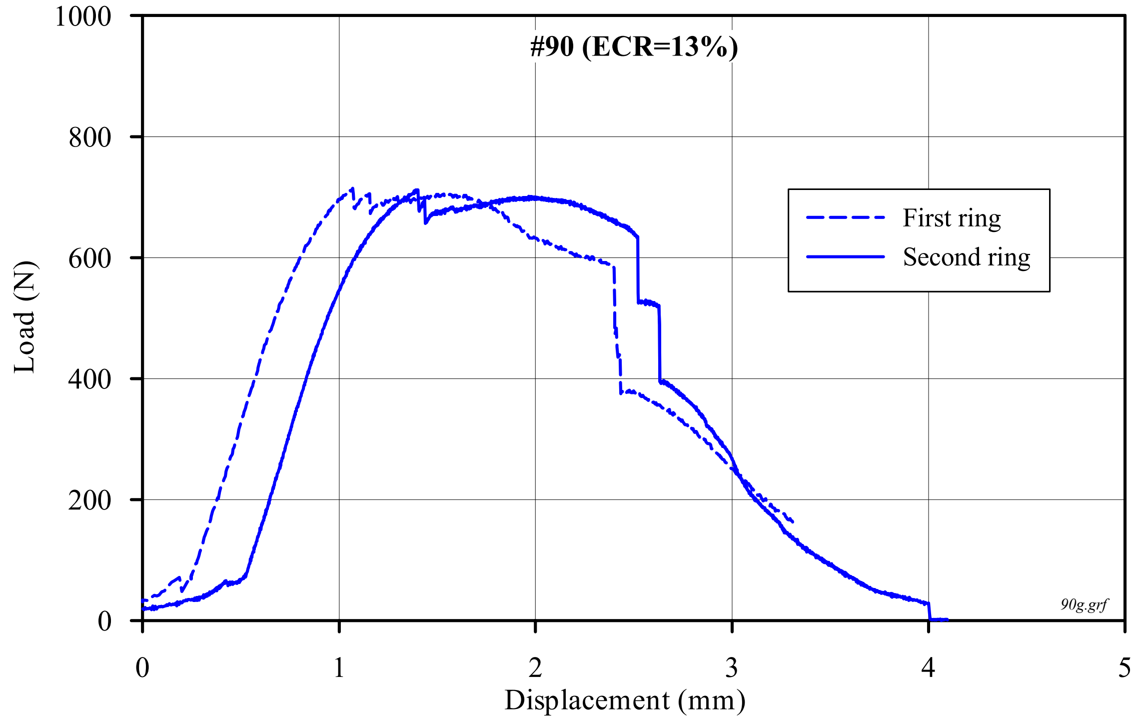
**The E110 cladding fabricated on the basis of the sponge Zr (G110-f) and oxidized at 1100 C demonstrates the same behavior as the Zry-4 cladding (no breakaway effect, a high margin of residual ductility)**

# Bulk Effect Studies. Zr Ingot Variations
## Additional information to verify the conclusions on G110-f mechanical behavior

**Results of ring compression tests**

↓

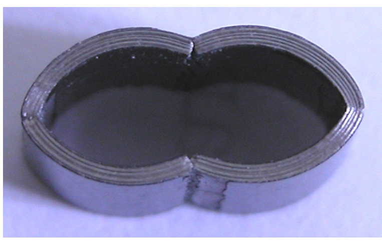
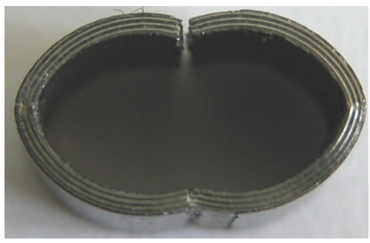
**Load-displacement diagrams for two rings from the sample #90 (13% ECR)**




↓

**Appearance of ring samples after mechanical tests**

↓

| #89 | #90 |
|---|---|
|  |  |

↓

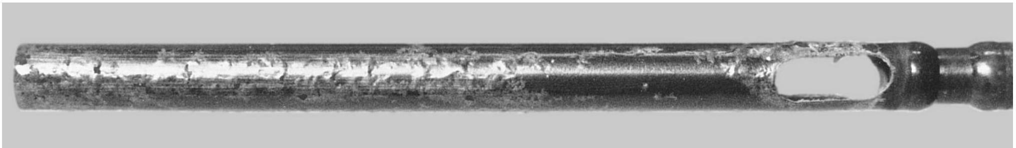
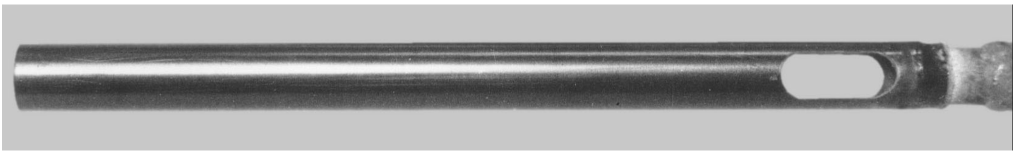
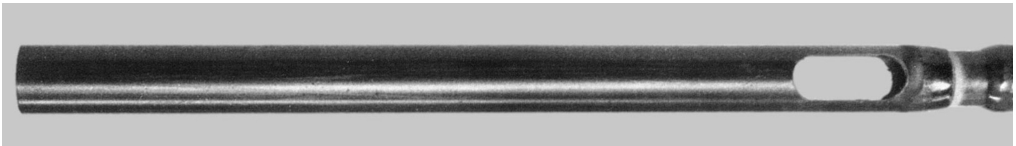
**Presented data confirm that the plastic deformation of G110-f oxidized samples was observed before the failure**

# Bulk Effect Studies. Zr Ingot Variations

## Test results with the G110-3ru cladding (E110 alloy on the basis Russian sponge Zr)

**1100 C, double-sided oxidation, 11.6–16.7% ECR**

↓

| Appearance of the claddings after the oxidation tests | |
|---|---|
| E110(standard), #96, 9.8% ECR $H_2 \rightarrow 4$ ppm |  |
| G110-3ru, #95, 11.6% ECR $H_2 \rightarrow 17$ ppm |  |
| G110-3ru, #97, 16.7% ECR |  |

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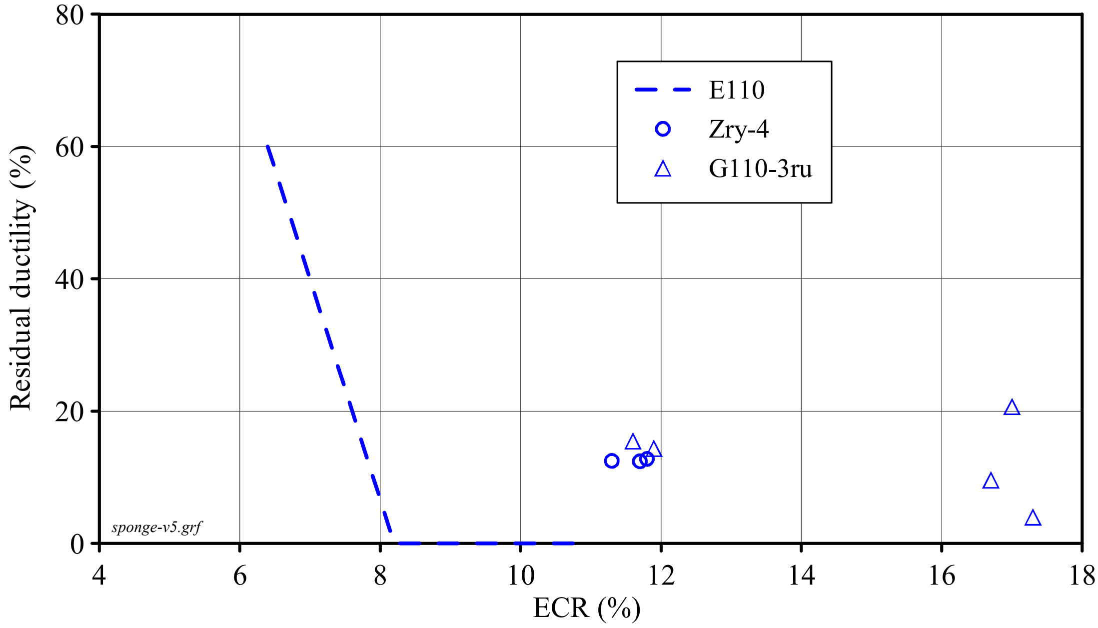



↓

**Conclusion:**

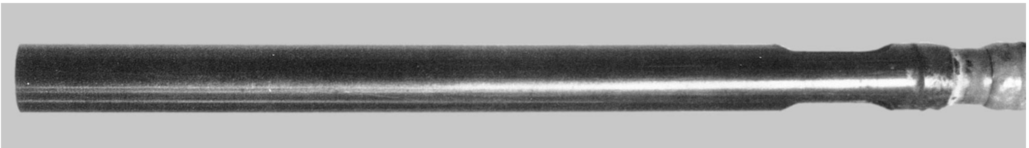
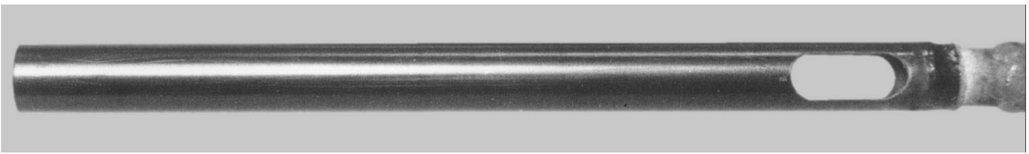
**The E110 cladding fabricated on the basis of Russian sponge Zr demonstrates the same (or better) behavior as the Zry-4 cladding does at the oxidation at 1100 C**

# Bulk Effect Studies. Zr Ingot Variations

## Comparative test results with G110-3ru and G110-3f (E110 alloy on the basis of French sponge Zr, iodide Zr, recycled scrap) claddings

**1100 C, double-sided oxidation, 11.5% ECR**

↓

| Appearance of the claddings after the oxidation tests | |
|---|---|
| G110-3f, #99, 11.5% ECR $H_2 \rightarrow 13$ ppm |  |
| G110-3ru, #95, 11.6% ECR $H_2 \rightarrow 4$ ppm |  |

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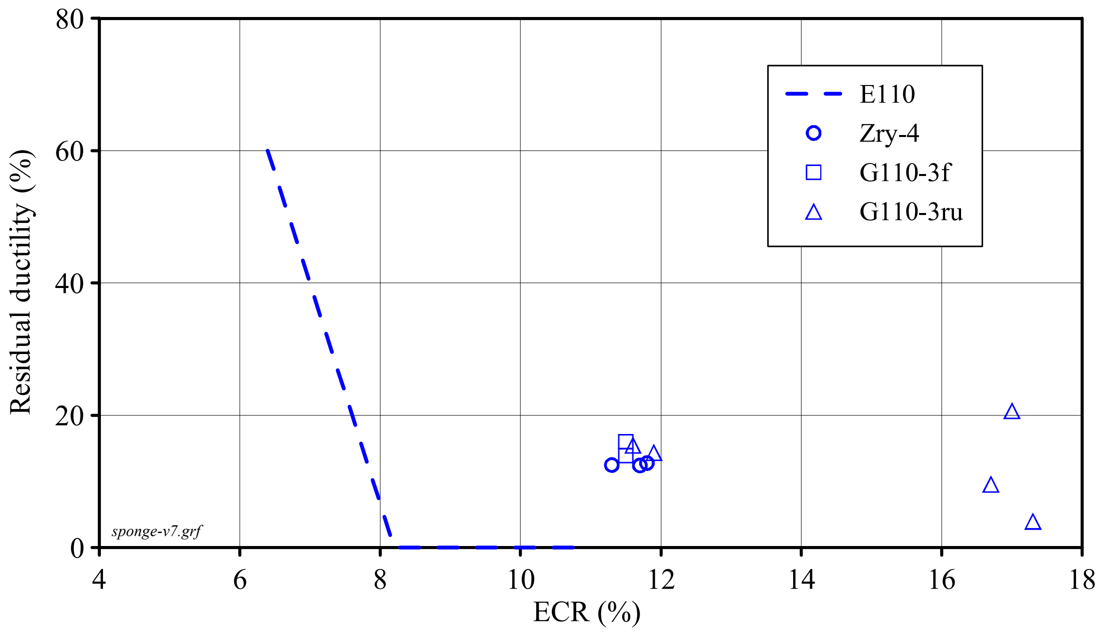



↓

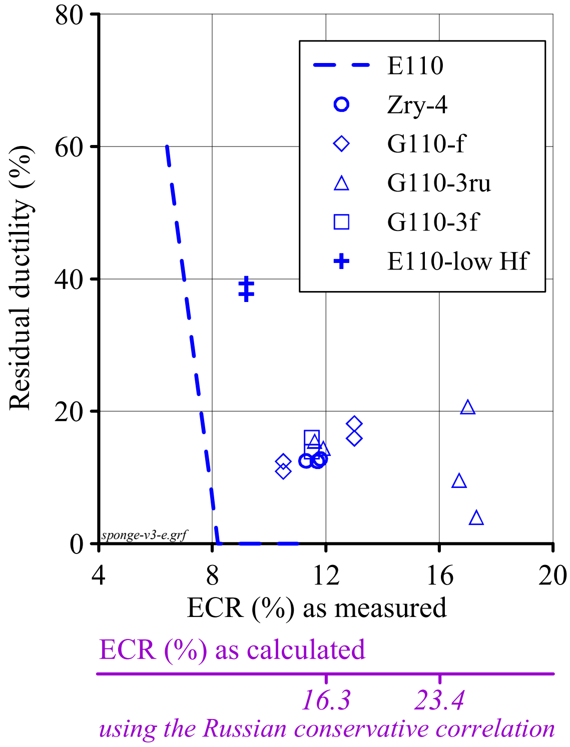
**Conclusion:**

**The oxidation behavior and residual ductility of E110 claddings of G110-3f and G110-3ru types are approximately the same**

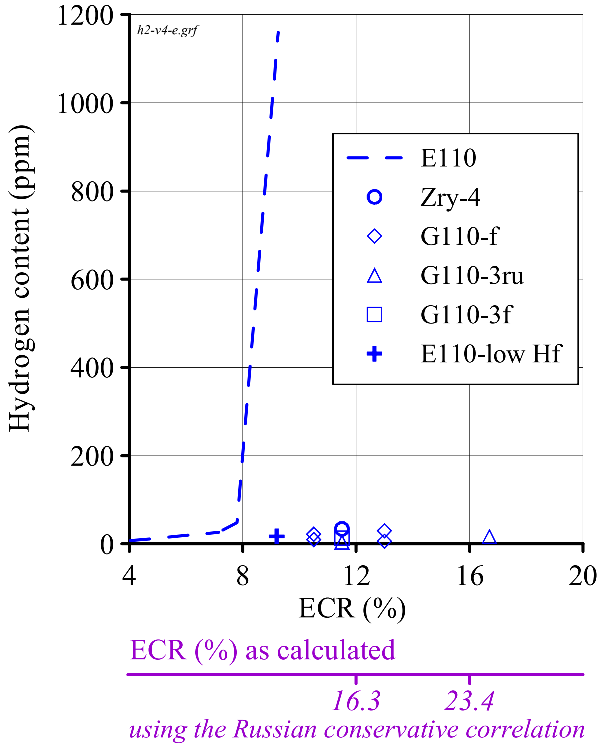
# Bulk Effect Studies. Zr Ingot Variations

## Summary of test results for the double-sided oxidation at 1100 C

### Residual ductility vs ECR




### Hydrogen content vs ECR




**General assumption laid in the basis for the development of conclusions: the only parameter (Zr ingot composition) was varied in these bulk effects studies. Other fabrication procedures were the same for the whole set of tested claddings**

↓

## Conclusions:

1. All claddings manufactured without electrolytic Zr (as the component of Zr ingot) have demonstrated a high margin of residual ductility up to 17% ECR as measured (25% ECR as calculated)
2. A typical tetragonal oxide was formed on the cladding surface
3. Hydrogen concentration in the cladding was very low
4. The test with the E110-low Hf cladding has shown the encouraging result also. But an additional test should be performed at the higher ECRs for the final conclusions on this cladding type

## Bulk Effect Studies. Zr Ingot Variations

## The purpose and results of additional oxidation and mechanical tests performed with E110 claddings manufactured on the basis of the sponge Zr

**Background of the problem**

It is known that E110 claddings demonstrate the worst oxidation behavior at the temperature around of 1000 C

↓

**Purpose of additional tests**

To verify the oxidation and mechanical behavior of G110-f and G110-3ru cladding under the worst temperature conditions

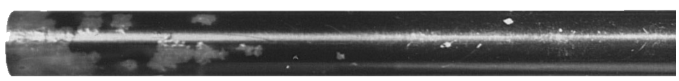
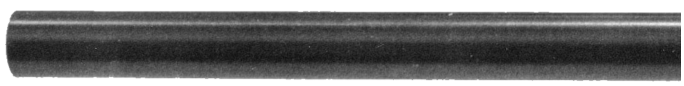
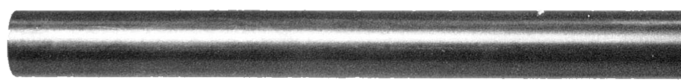
# Bulk Effect Studies. Zr Ingot Variations

## Comparative data on the behavior of E110, G110-f, G110-3ru claddings at the temperature 1000 C

**1000 C, double-sided oxidation, 7→9% ECR**

↓

**Appearance of claddings after oxidation**

| | | | |
|---|---|---|---|
| #44 | E110<br>ECR=7.7% | $t_{ef}$=865 s |  |
| #91 | G110-f<br>ECR=6.5%<br>($H_2$→28 ppm) | $t_{ef}$=2016 s |  |
| #93 | G110-f<br>ECR=8.5%<br>($H_2$→12 ppm) | $t_{ef}$=5013 s |  |
| #98 | G110-3ru<br>ECR=6.9%<br>($H_2$→16 ppm) | $t_{ef}$=2519 s |  |
| #101 | G110-3ru<br>ECR=8.9%<br>($H_2$→11 ppm) | $t_{ef}$=5028 s |  |

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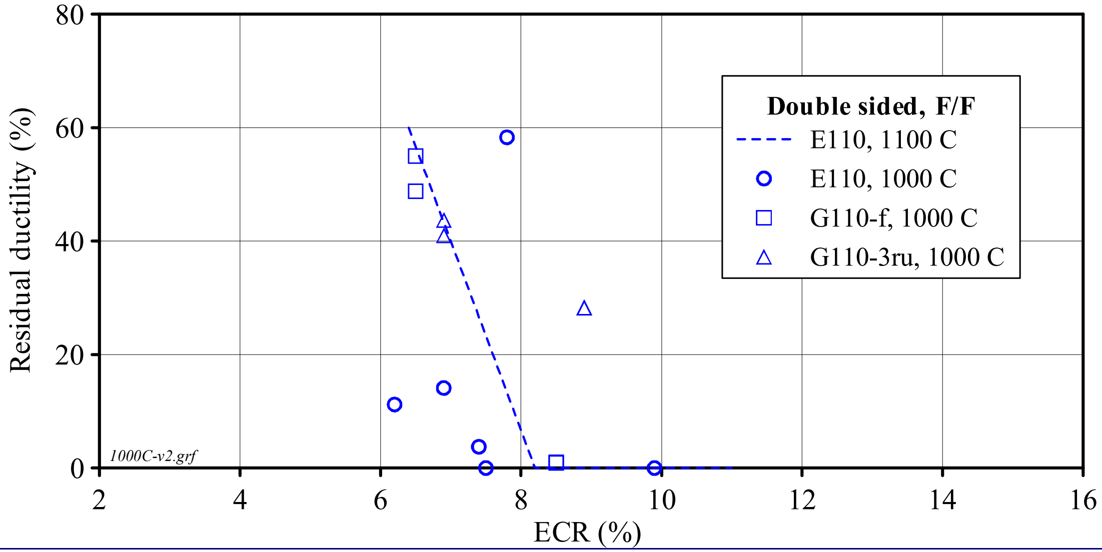
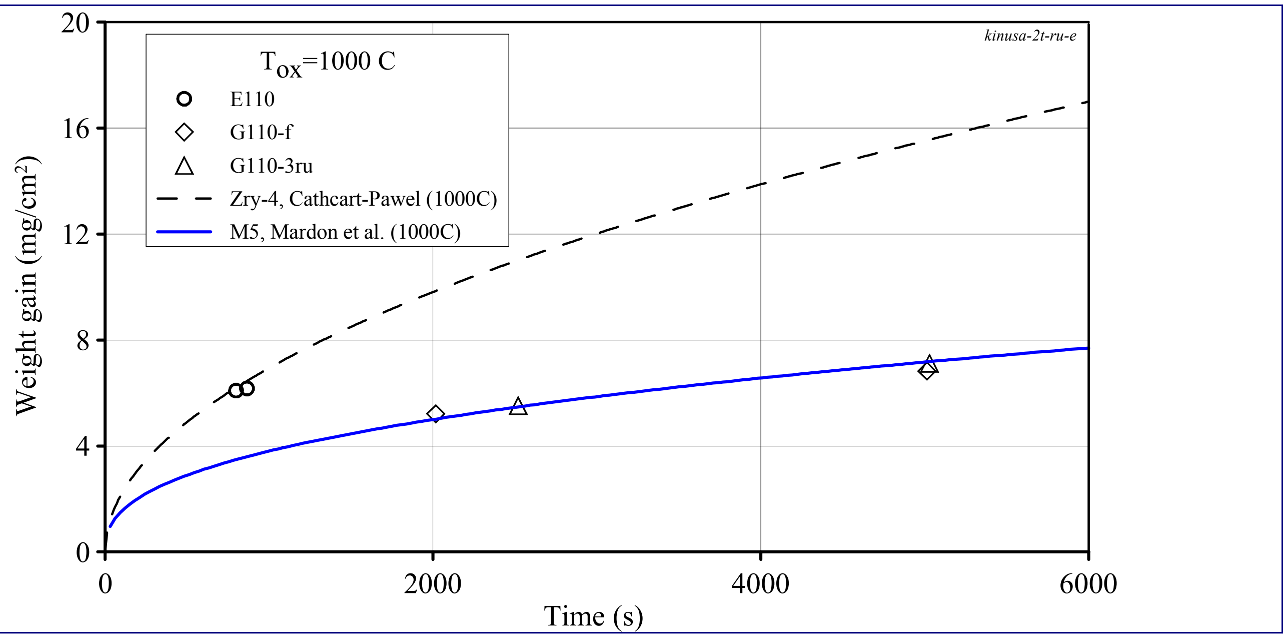
**Residual ductility of E110, G110-f, G110-3ru claddings as a function of the ECR at 1000 C**



# Bulk Effect Studies. Zr Ingot Variations

## Comparative data on the behavior of E110, G110-f, G110-3ru claddings at the temperature 1000 C




↓

**Conclusions:**

1. The following general differences in the oxidation behavior of the E110 cladding and G110-f, G110-3ru claddings were revealed:

- ~800 s of the oxidation are needed to achieve the zero ductility threshold for the E110 standard cladding
- ~5000 s of the oxidation are needed to achieve the zero ductility threshold for the G110-f cladding
- ~5000 s of the oxidation do not allow to achieve the zero ductility threshold for the G110-3ru cladding

2. The G110-f, G110-3ru and M5 (in accordance with published data) claddings have the same oxidation kinetics. Moreover, the oxidation rate of these claddings is much less than that for the Zry-4 cladding
3. Such durations of the oxidation as 2000 s, 5000 s are outside of the practical interest for the large break LOCA safety analysis
4. In this context, the question about the representativity of the ECR as the universal safety criterion could be formulated

# Bulk Effect Studies.
## Variations of cladding fabrication process

**Major provisions of the approach used for the analysis of the relationship between the cladding fabrication process and the cladding oxidation behavior**

↓

1. The analysis of the numerous studies performed in this field during the last thirty years allows to conclude that all differences in fabrication procedures, which are important for the oxidation behavior of any of zirconium-niobium alloys are reflected in the cladding microstructure
2. In accordance with results of these investigations, the major goal for the development and improvement of the fabrication process of zirconium-niobium alloys is to develop the following parameters of the microstructure:

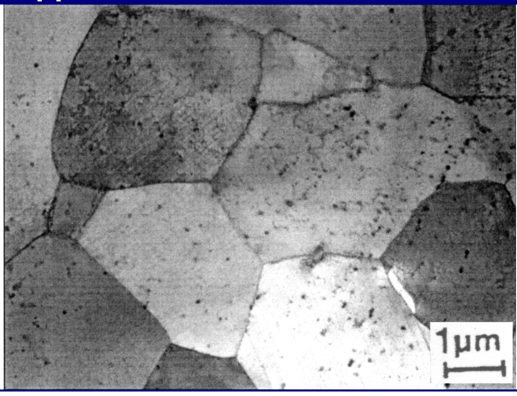
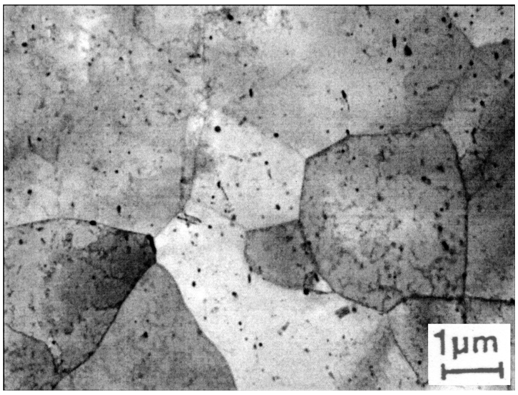
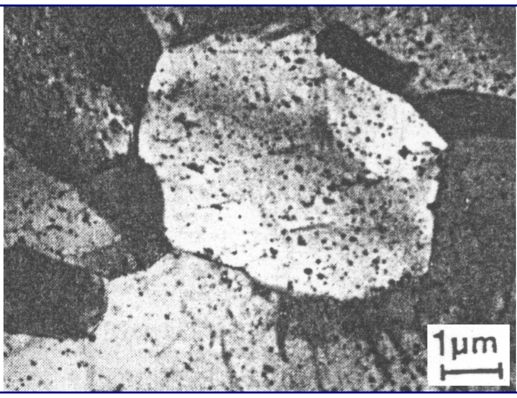
- maximum degree of the recrystallization
- small size of $\alpha$-Zr grain
- uniform distribution of secondary precipitates inside the Zr matrix
- small size and high density of distribution of globular niobium precipitates

3. Taking into account these findings, it was decided to estimate the dependence of the oxidation behavior of these alloys on the fabrication procedures using the comparison of the TEM data characterizing the microstructure of each of alloys. To obtain these data, special TEM examinations were performed with E110, G110-f, G110-3ru, G110-3f, E110-low Hf cladding materials

# Bulk Effect Studies.
## Variations of cladding fabrication process

The attempt to reveal possible differences in the cladding fabrication process using the results of TEM examinations

↓

<table>
<tr><th>Type of cladding</th><th>Appearance of the microstructure</th><th>Characterization of the microstructure</th></tr>
<tr><td>E110 standard Russian as-received tubing</td><td></td><td>♦ fully recrystallized structure (αZr matrix with βNb globular precipitates)<br>♦ grain size 2.8 µm<br>♦ a high level of dispersion of β-Nb precipitates:<br>D = 45, 60 nm (results of measurements in two laboratories)<br>$N = 1.84 \times 10^{14}$ cm$^{-3}$</td></tr>
<tr><td>G110-f as-received tubing fabricated with French sponge Zr by the Russian procedure</td><td></td><td>♦ fully recrystallized structure (αZr matrix with βNb globular precipitates and $Zr(Nb, Fe)_2$ precipitates)<br>♦ grain size 3.2 µm<br>♦ a high level of dispersion of precipitates<br>♦ parameters of β-Nb precipitates:<br>D = 41–43 nm (results of measurements in two laboratories)<br>$N = 1.8 \times 10^{14}$ cm$^{-3}$</td></tr>
<tr><td>M5 as-received tubing<br>(All presented data were taken from D.Gilbon et.al. "Irradiation Creep and Growth Behavior, and Microstructural Evolution of Advanced Zr-Base Alloys", ASTM-STP1354</td><td></td><td>"Thermodynamically stable microstructure is characterized by a highly refined dispersion of β-Nb precipitates (D = 45 nm, $N = 1.5 \times 10^{14}$ cm$^{-3}$) with no alignment of particles"</td></tr>
</table>

↓

**Preliminary conclusions:**

1. General differences in microstructures of studied alloys (E110, G110-f, G110-3ru, G110-3f, E110-low Hf) were not revealed
2. Appearances and parameters of microstructures of these cladding materials are similar to these for M5 alloy
3. It can be assumed that differences in the oxidation behavior and embrittlement thresholds of these alloys are not a direct function of fabrication process

# Conclusions

1. Experimental studies (performed during 2001–2002 with Russian Zr-1%Nb claddings manufactured from the E110 alloy) allowed to reveal that the nodular corrosion accompanied by hydrogen absorption is responsible for the earlier embrittlement of E110 cladding in comparison with Zry-4 cladding under LOCA relevant conditions
2. The scientific discussion conducted on the basis of the analysis of results from this stage of the work with the participation of specialists from JSC "TVEL", VNIINM (Bochvar Institute, Russia), ANL, NRC (USA), IRSN (France) became the basis for the development of a coordinated program of investigations performed at RRC KI/RIAR and ANL during this year
3. The following phenomena were selected for these studies:

- surface effects and effect of geometrical sizes
- bulk effects connected with the chemical composition of impurities in the cladding material
- bulk effects connected with variations of the fabrication process

4. The research performed by RRC KI/RIAR specialists allowed to conclude the following:

- different procedures for the cladding surface conditioning lead to the different oxidation behavior of the E110 cladding; surface polishing improves the oxidation behavior, but the etching leads to the negative results
- preliminary analysis of the data base obtained to determine the dependence of post-LOCA ductility as a function of current cladding fabrication processes has shown that this is not the key reason for different behavior of zirconium-niobium alloys
- in accordance with current data, the chemical composition of impurities in the cladding material is considered as a key factor responsible for different oxidation behavior of different zirconium-niobium alloys
- special tests performed with experimental types of the E110 cladding manufactured using different methods of Zr ingot preparation have shown that the oxidation behavior and embrittlement threshold of these cladding are the same as these for Zry-4 and M5 claddings

# Recent data on M5™ Alloy under LOCA Conditions (as compared to Zy-4 behavior)

Nicolas WAECKEL (***EDF***, France)

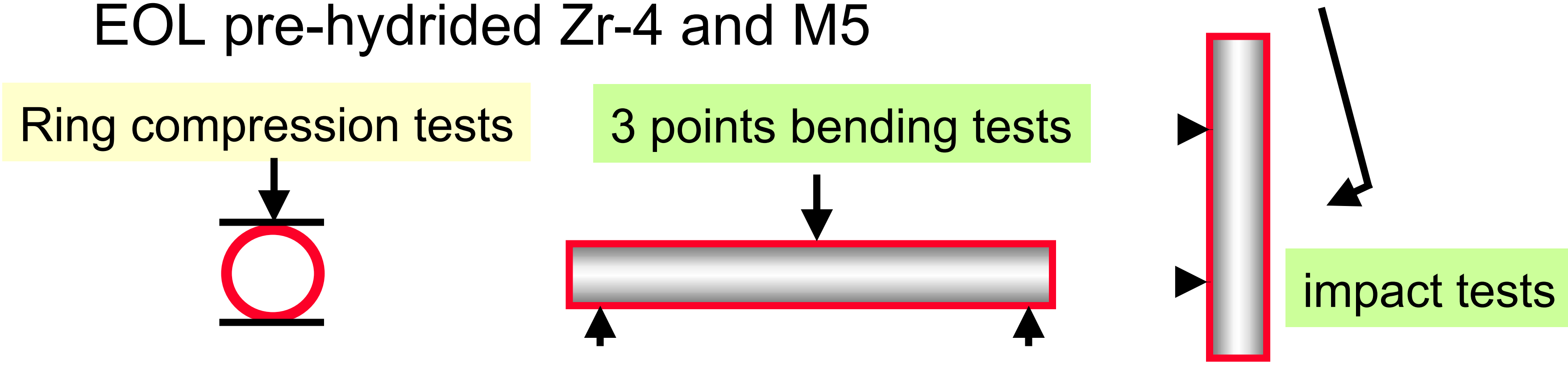
Jean-Paul MARDON (**Framatome ANP**, France)

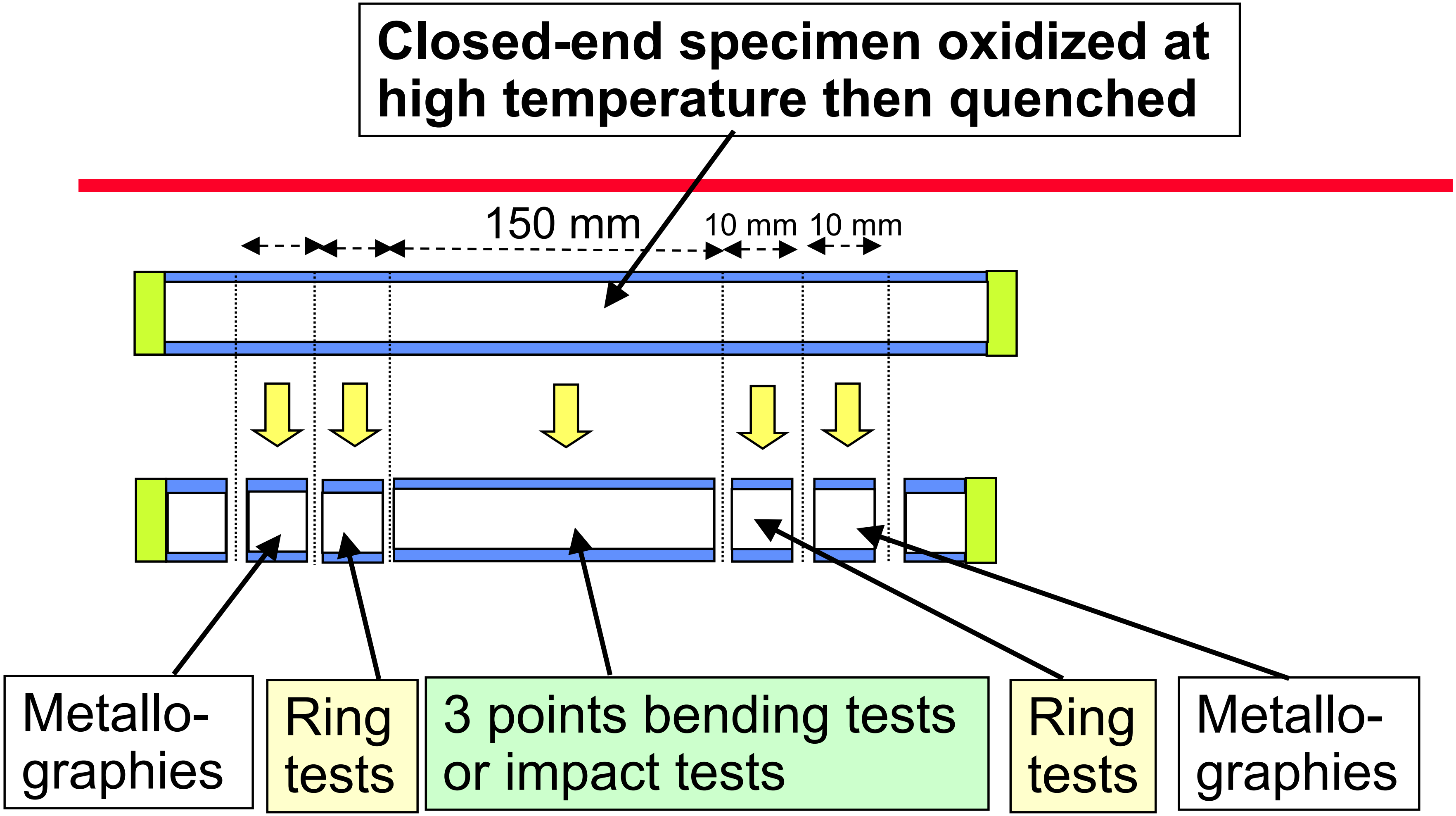
# How does Nb-based alloys compare to Zr-4 under LOCA conditions ?

- Recent papers of experimental results on E110 cladding material suggested that all **Nb-based alloys may behave worse than Zr alloys** under LOCA conditions
  - Higher H pick up during high temperature oxidation
  - Lower post-quench residual ductility
- The purpose of this paper is to demonstrate that the Nb-based Alloy-**M5™** manufactured by Framatome - behaves similar to (or better than) **Zr-4** under prototypical LOCA conditions

# Current LOCA limits are based on post quench mechanical tests

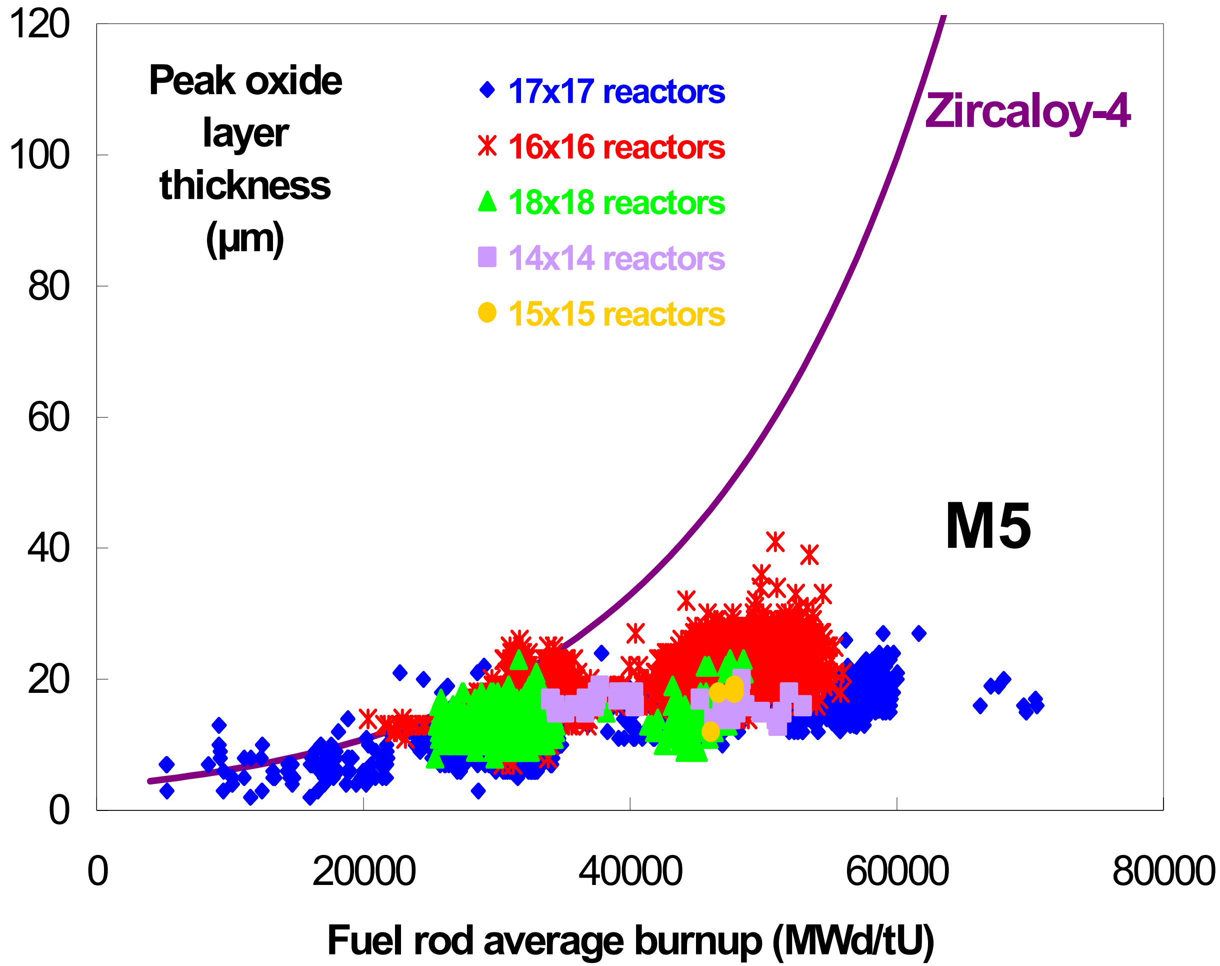
- The LOCA limits are somewhat related to the **zero residual ductility** limit of the cladding measured **after** the LOCA transient, **at low temperature** using ring compression tests (Hobson 1970) or impact tests (Chung 1973)
  - »LOCA limits : ECR < 17% and PCT < 1204 $^{o}$C
- To adopt the 1972 approach, *EDF*, Framatome-ANP and CEA and have performed a series of **post-quench mechanical tests** with as–received and prototypical EOL pre-hydrided Zr-4 and M5



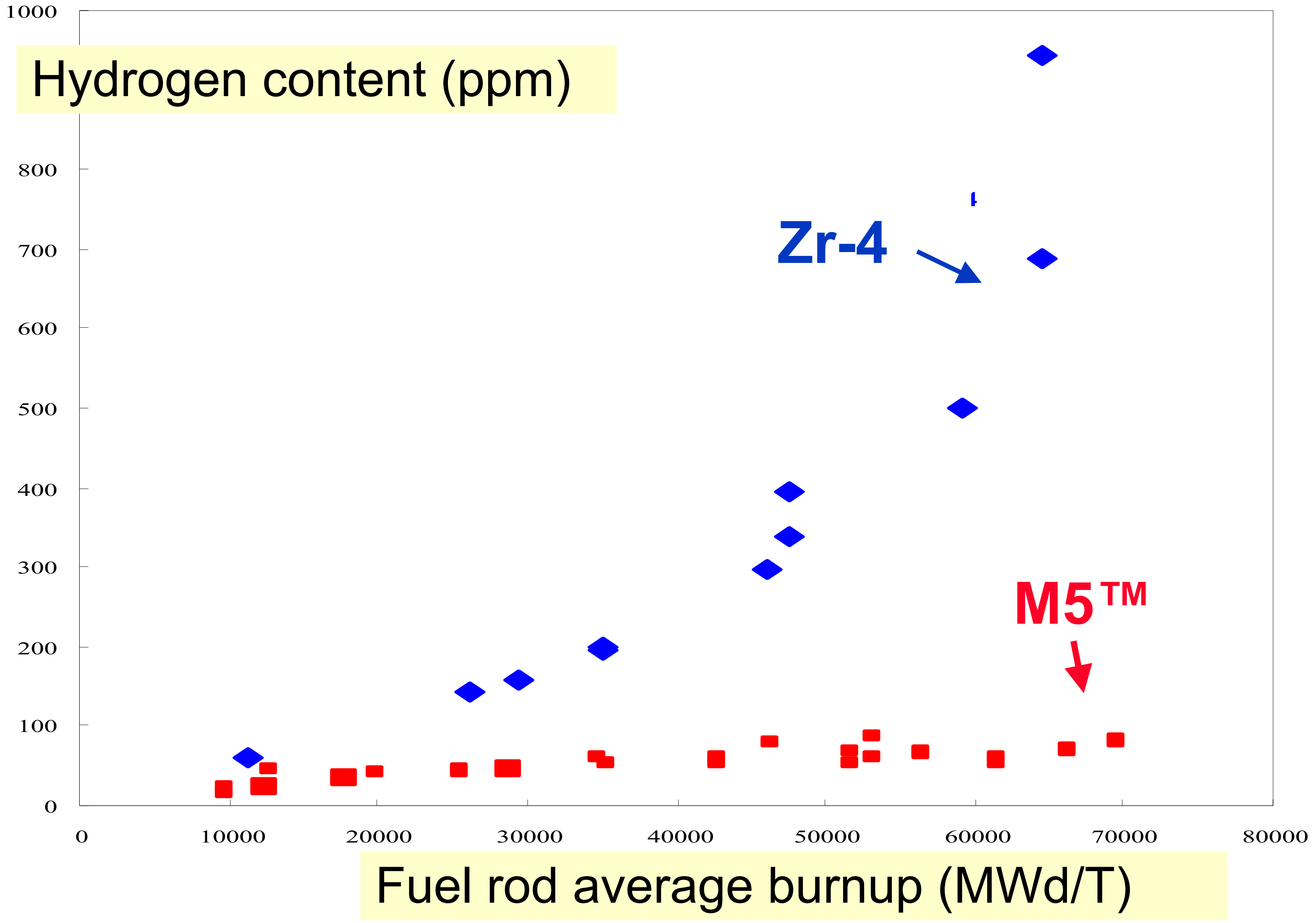



- 4 levels of oxidation temperatures : 1000 –1100 –1200 -1300ºC
- 4 levels of ECR
- Tests on **pre-hydrided** specimens (to simulate in-reactor corrosion)

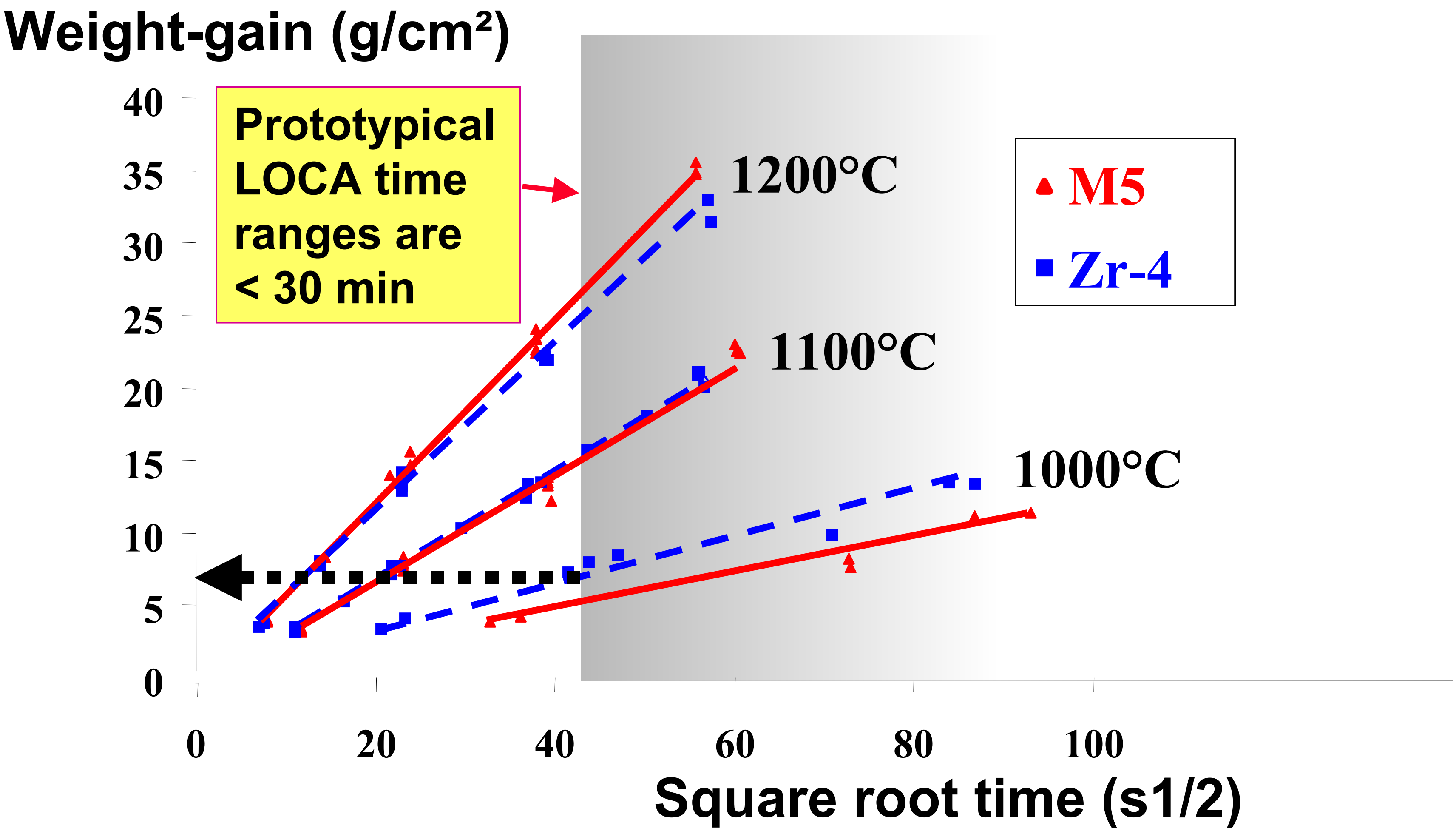
# M5™ exhibits a much better in-reactor behavior than Zr-4



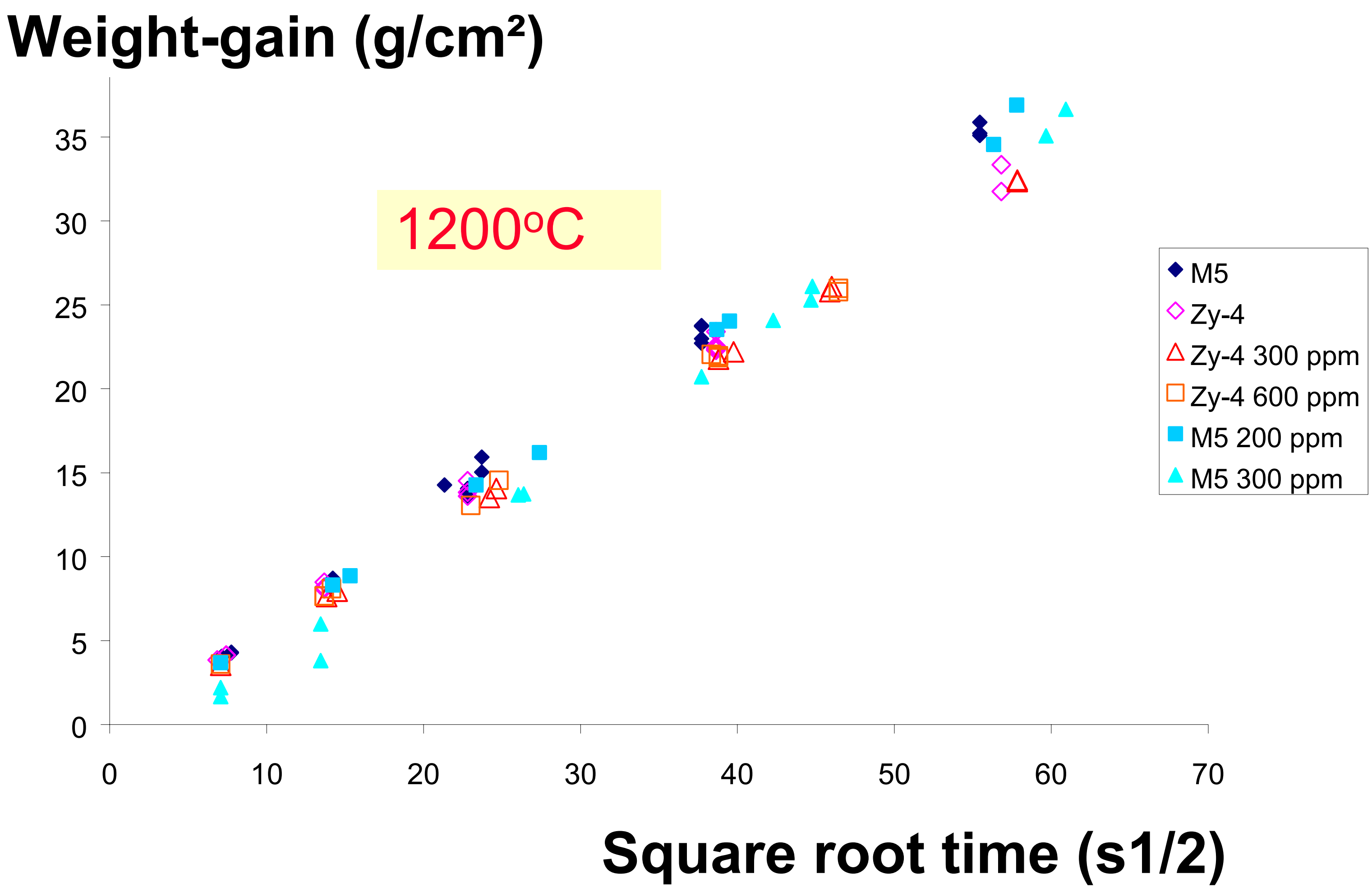
# Hydrogen Uptake of M5™ is extremely low



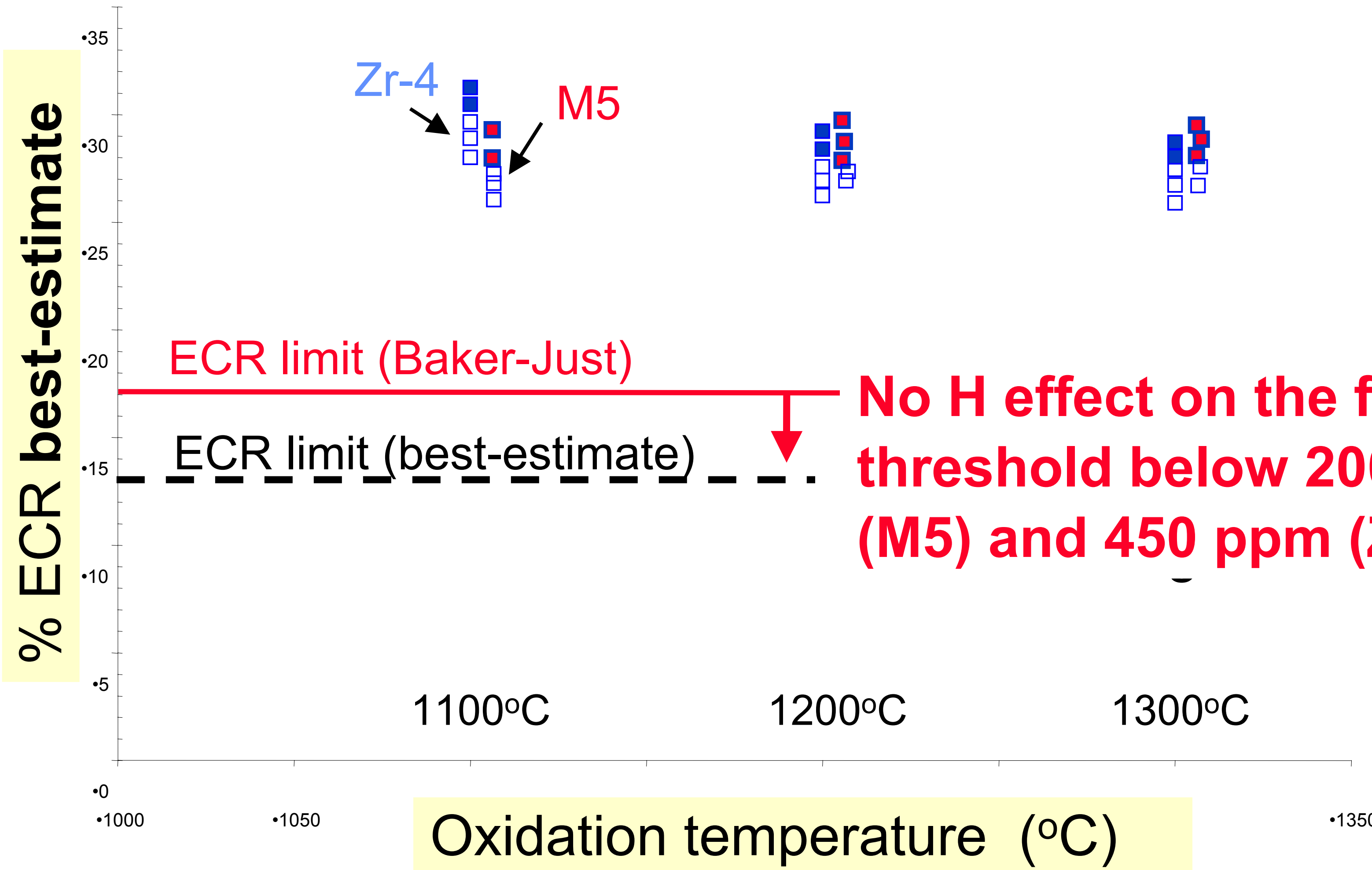
# M5™ oxidation kinetics are equivalent to Zr-4



# No impact of H on oxidation kinetics for both alloys



# “Failure” upon quench is the same for Zr-4 and M5™ -

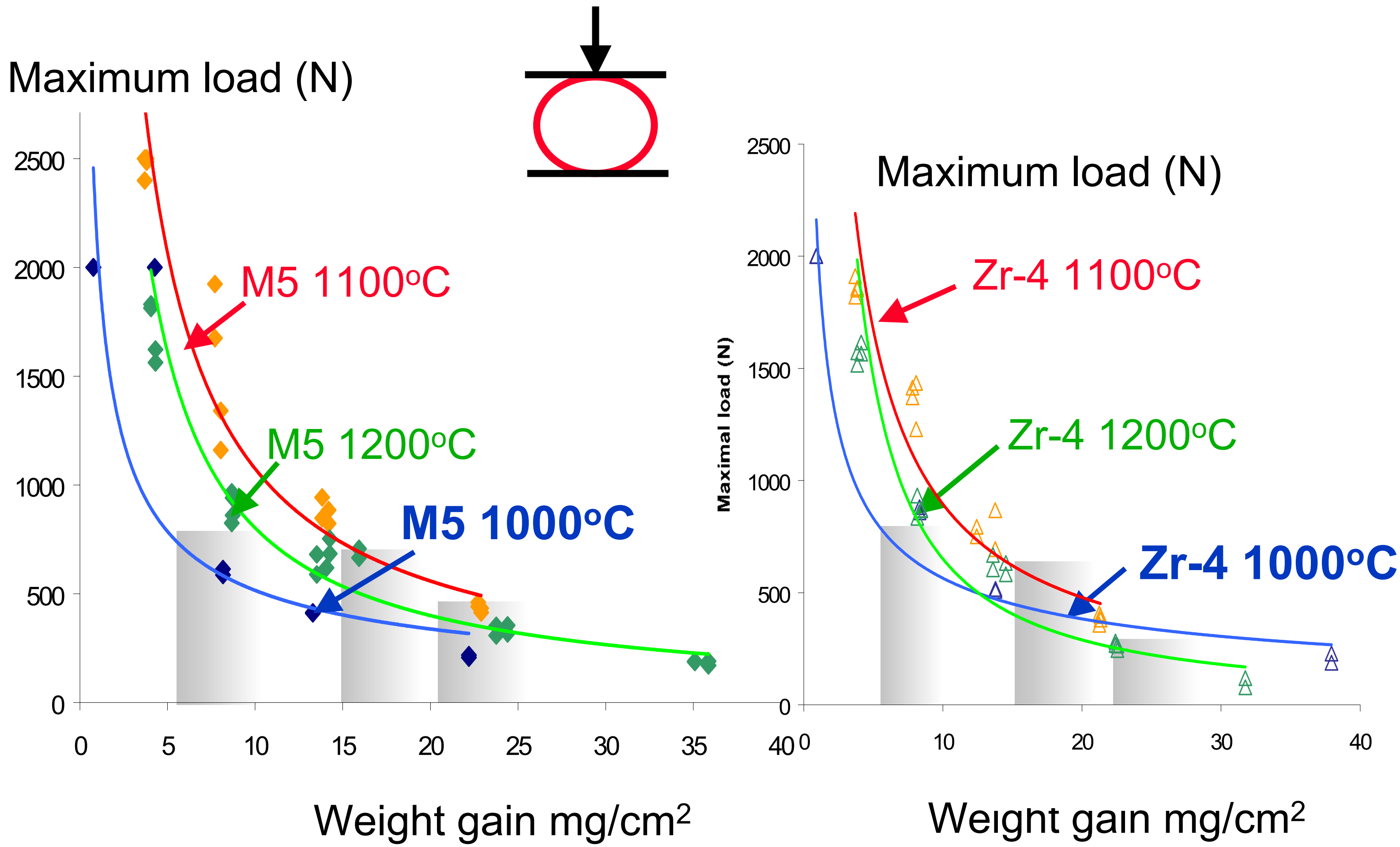



**No H effect on the failure threshold below 200ppm (M5) and 450 ppm (Zr-4)**

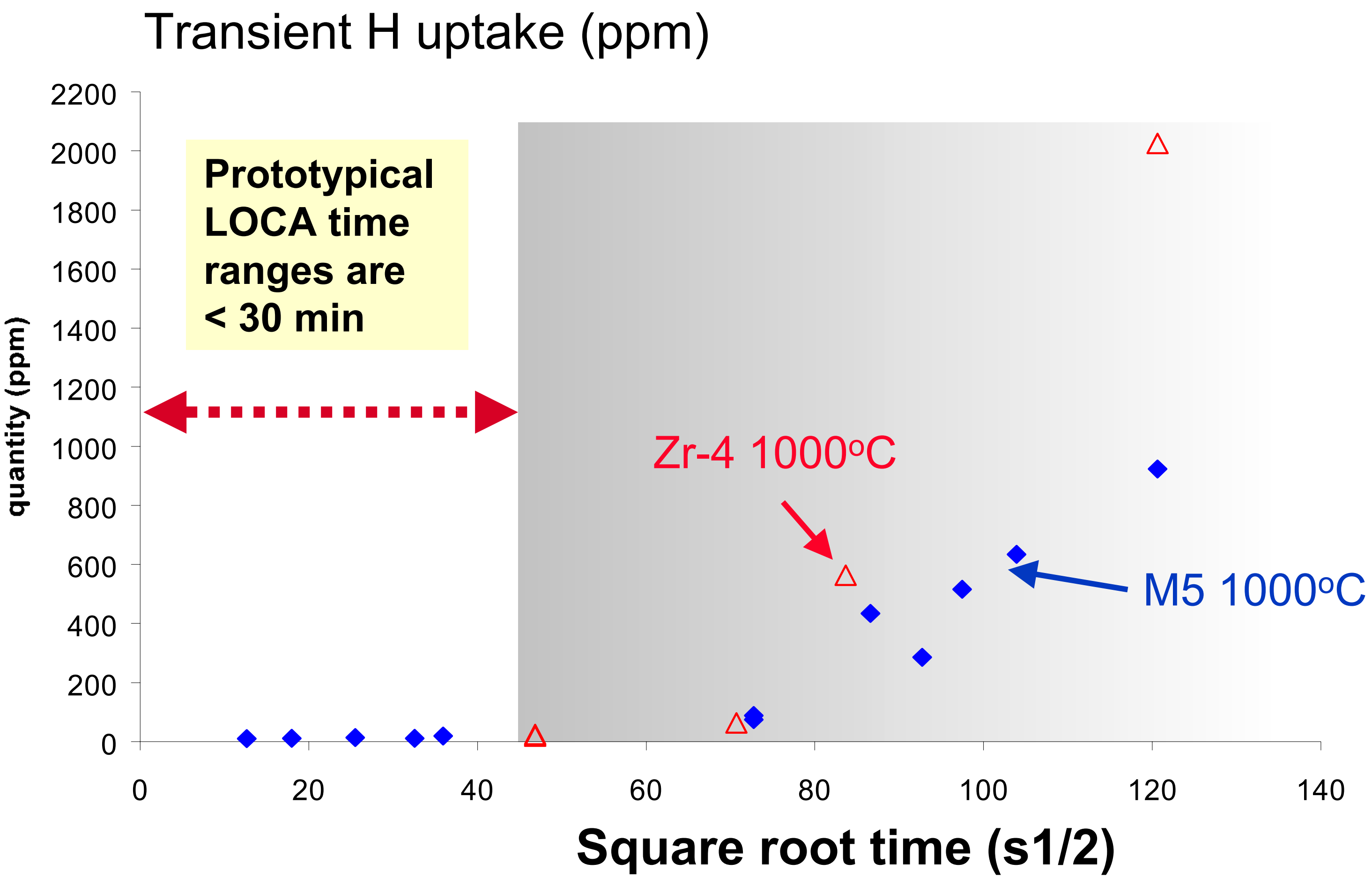
# Behavior upon quench

- High temperature oxidation kinetics and quenching behavior are **equivalent for M5 & Zr-4**
- The failure thresholds of **pre-hydrided** M5™ and Zr-4 remain unchanged (**below 200ppm (M5) and 450 ppm (Zr-4))**
- Negligible **transient hydriding** was observed after oxidation and quench (< 25 ppm)
- Time to failure upon quench at **1000°C** for fresh and pre-hydrided specimens are much longer than the time envisioned for LOCA

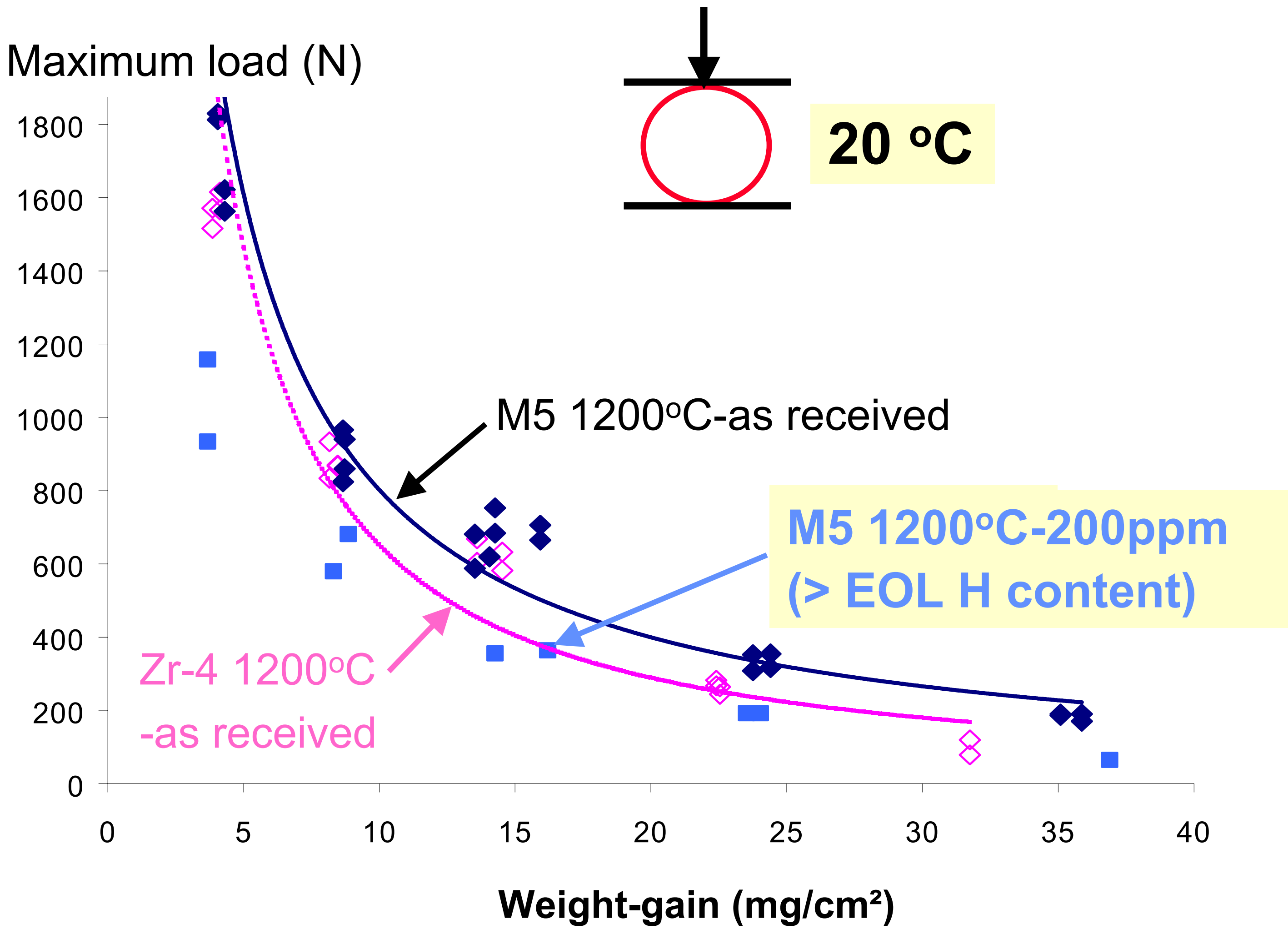
# Post-Quench Ring Compression Tests at 20°C



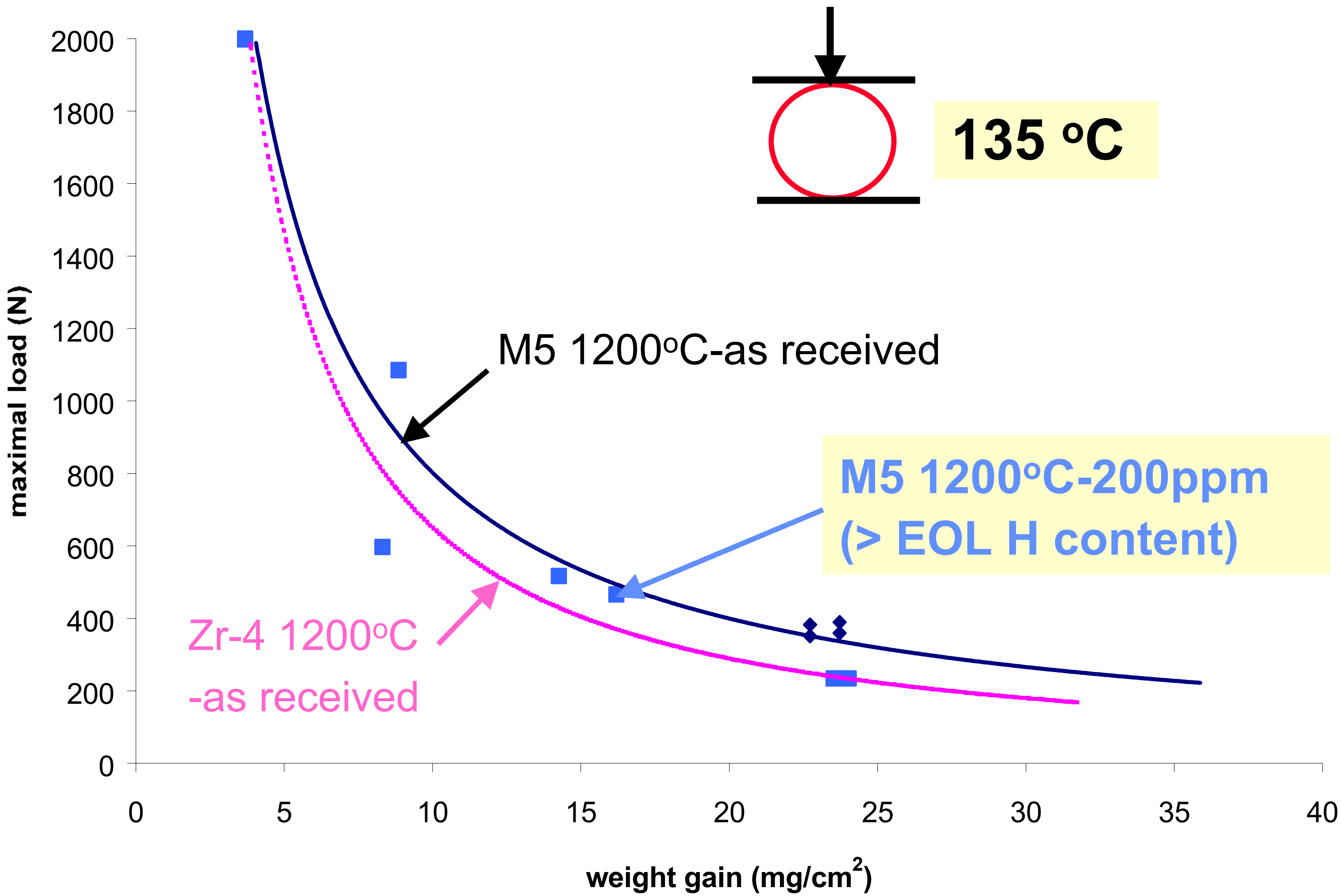
# At 1000°C M5™ break-away H uptake is equivalent to Zr-4



# Post-quench RT ring compression tests on M5™ and Zr-4



# At 135 °C the post-quench residual ductility of pre-hydrided M5™ is restored



# Post-quench ductility

- **No** runaway oxidation and related embrittlement was observed up to 1400°C during prototypical LOCA time frame (<30 min)

- **In-service oxidation/hydriding has NO or little effect** on the oxidation, the quench behavior or the post quench ductility of M5™

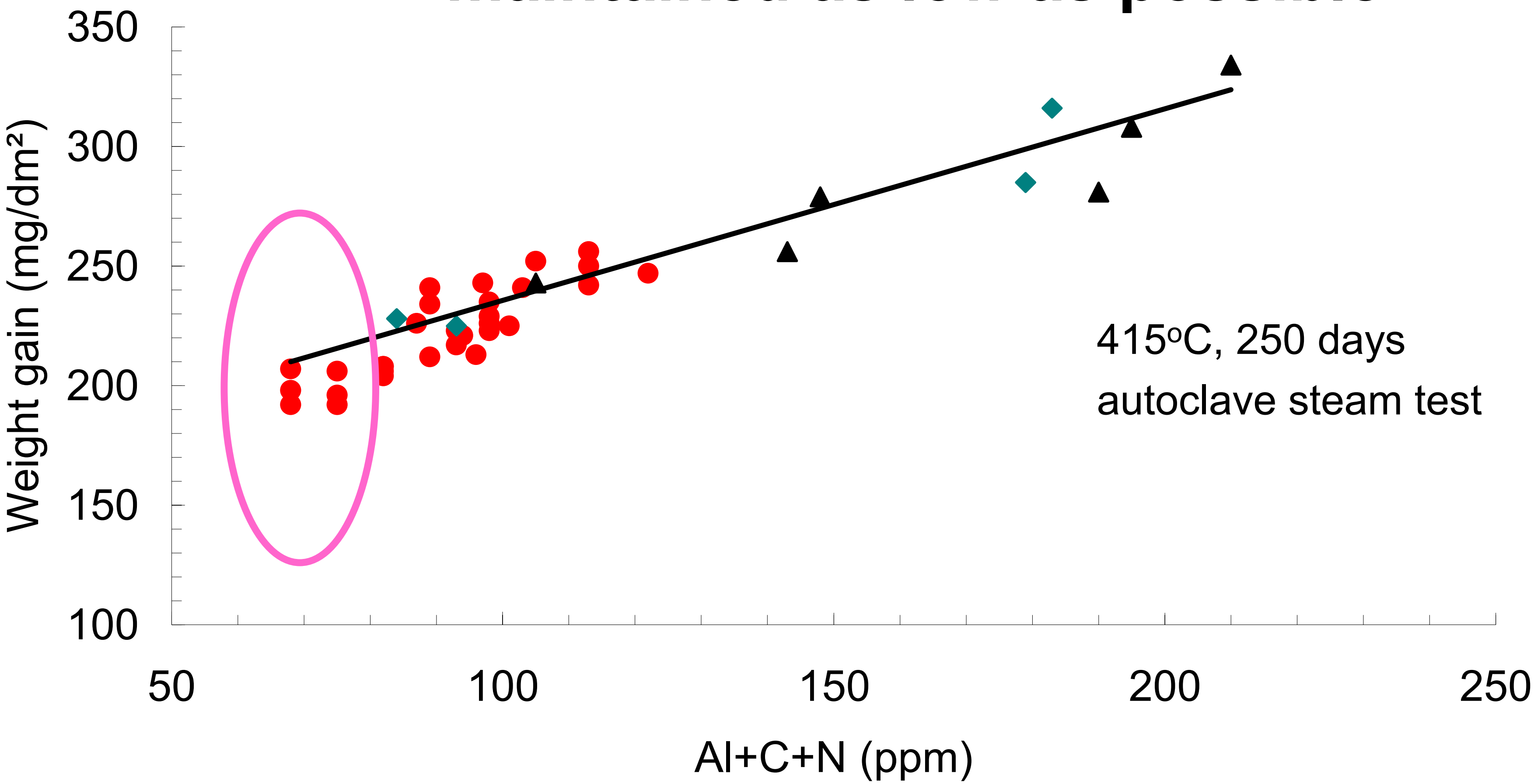
- Contrary to E110, the **post-quench mechanical properties** of M5™ are similar to (or better than) those of Zr-4

# Why M5™ is different from other Nb Alloys ?

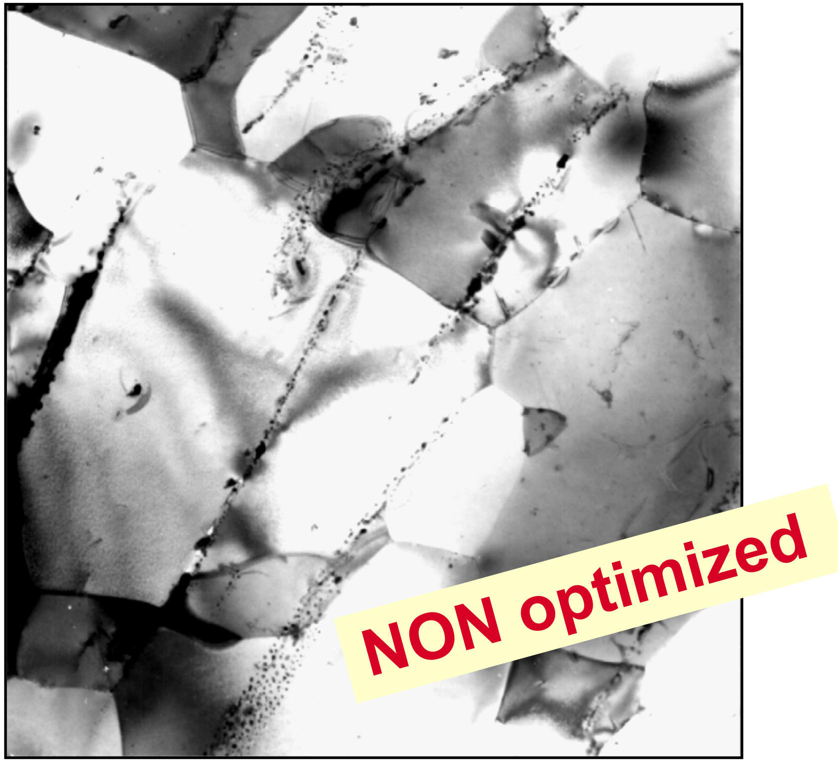
- Recent studies suggest impurities, SPPs size and distribution or the surface finish may impact the Nb based alloys behavior under LOCA conditions
- During the development of M5™, potential effect of many parameters have been investigated and optimized
  - **Chemical composition**
    - Alloying elements (Nb, Fe, Cr, Ni, V, O…)
    - Impurities (Ca, Mg, Sn, Si, Zn, Al, N, H, S …)
  - **Manufacturing process**
    - Annealing temperatures
    - Quenching modes
    - Number of meltings
    - Number of pilgering steps
    - Final annealing temperature
    - Surface finish

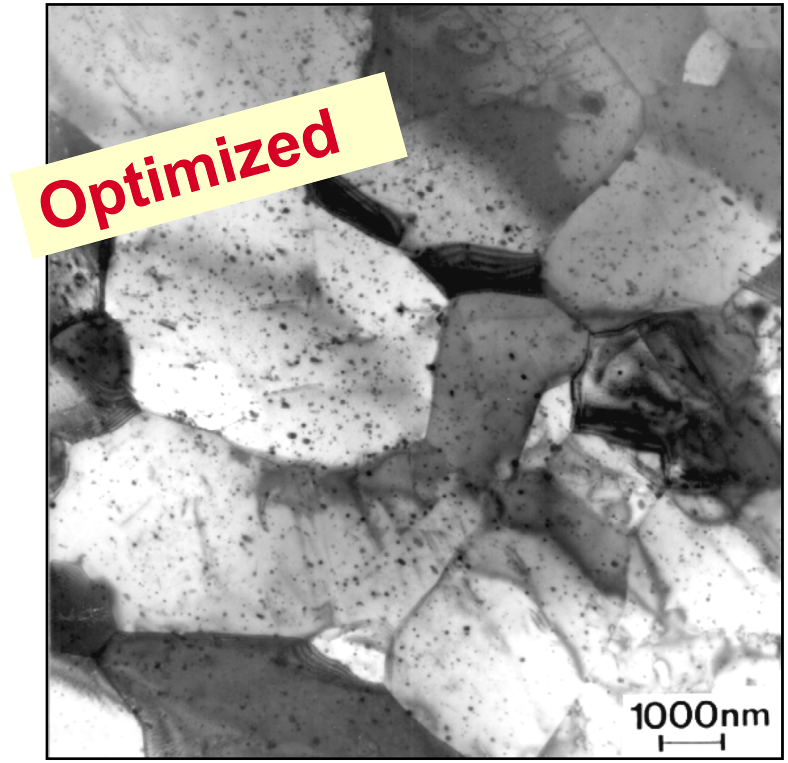
# Variations of the addition elements within the specified ranges have little impact…

## …except for Al-C-N that must be maintained as low as possible



# The Final Annealing Temperature has a major impact

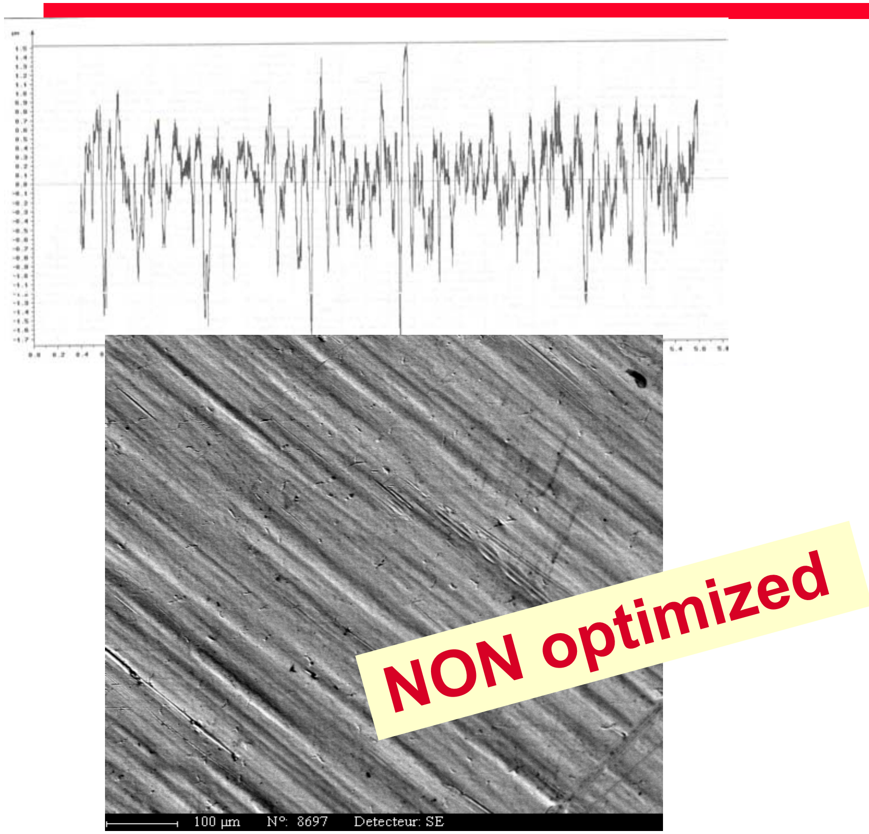


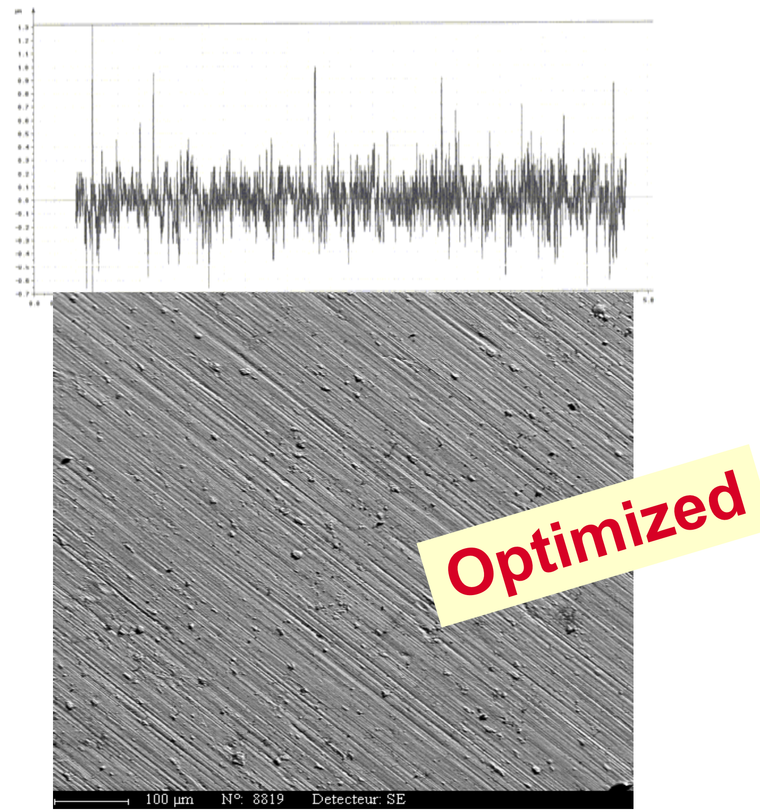




**High T process**
**(β-Zr, β-Nb) alignments // to the rolling direction**
**→heterogeneous corrosion**

**M5™ "low temperature" process**
**β-Nb uniformly distributed - No β-Zr**
**→ stable microstructure**

# OD surface finish strongly impacts the oxidation behavior




High OD surface roughness
→higher oxidation kinetics

**OD Standard M5$^{TM}$ process** : mechanical polishing →low surface roughness and low fluorine contamination→excellent oxidation behavior

# Conclusions

- Contrary to some Nb alloys, the **post-quench mechanical properties** of prototypically pre-hydrided M5™ are similar to (or better than) those of Zr-4

- M5™ behavior under LOCA conditions is **fairly robust** regarding
  - The chemical composition variations within the spec
    - Only Al+C+N has to be controlled and as low as possible
  - The manufacturing process variants
    - The last annealing temperature has to be lower than the eutectoid temperature and the OD surface has to be smooth

- The LOCA criteria (PCT<1204°C and ECR<17%) are fully justified and can be used for M5™ conservatively

# EFFECTS OF PELLET EXPANSION AND CLADDING HYDRIDES ON PCMI FAILURE OF HIGH BURNUP LWR FUEL DURING REACTIVITY TRANSIENTS

T. Fuketa, T. Sugiyama, T. Nakamura,
H. Sasajima and F. Nagase
Japan Atomic Energy Research Institute

October 21, 2003
Nuclear Safety Research Conference
Washington, DC, USA

# This presentation will cover;

- ✓ Recent results from pulse tests in the NSRR
  OI-10 and -11 with MDA and ZIRLO$^{TM}$, respectively
- ✓ Peak hoop strain at cladding failure in the NSRR experiments
- ✓ Tube burst test and ring-tensile test with un-irradiated, artificially-hydrided cladding
- ✓ Future NSRR experiments
  including tests with fuels shipped from Europe and with newly developed high-temperature capsule

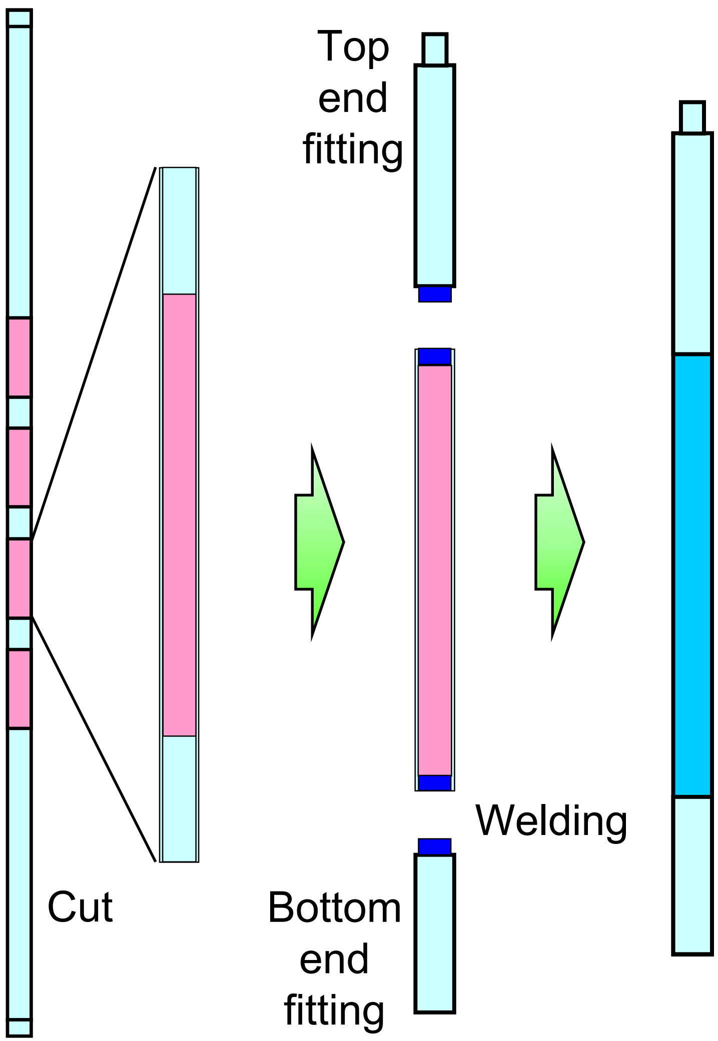
# NSRR pulse irradiation tests

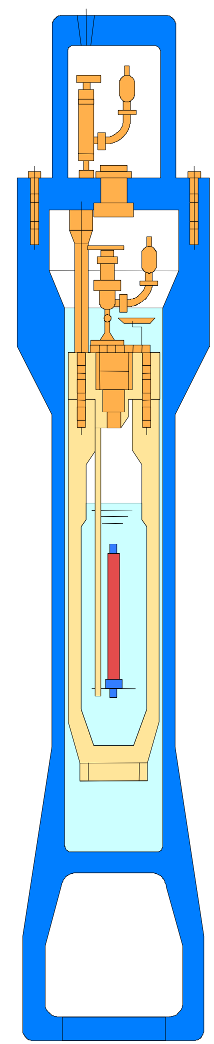
Pulse irradiation

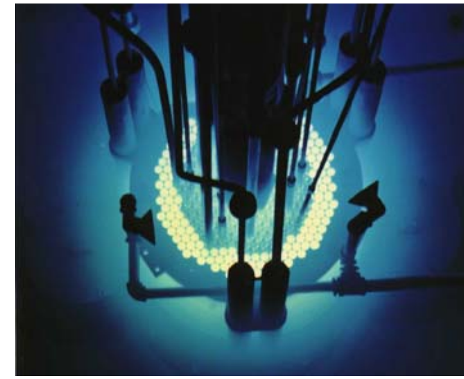
Irradiated rod from power plant ~4m

Test fuel rod
Total length: ~280 mm
Fuel stack: ~135 mm

Test capsule



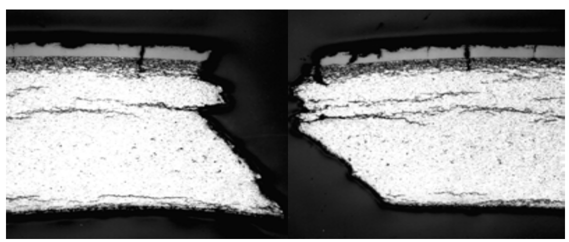


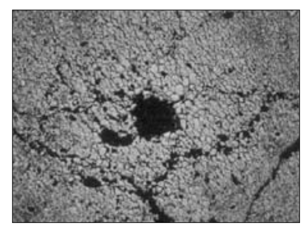
**Transient measurements:**

- Cladding surface temperature
- Coolant Water temperature
- Rod internal pressure
- Capsule internal pressure
- Fuel stack elongation
- Cladding elongation
- Cladding hoop strain
- Water column velocity

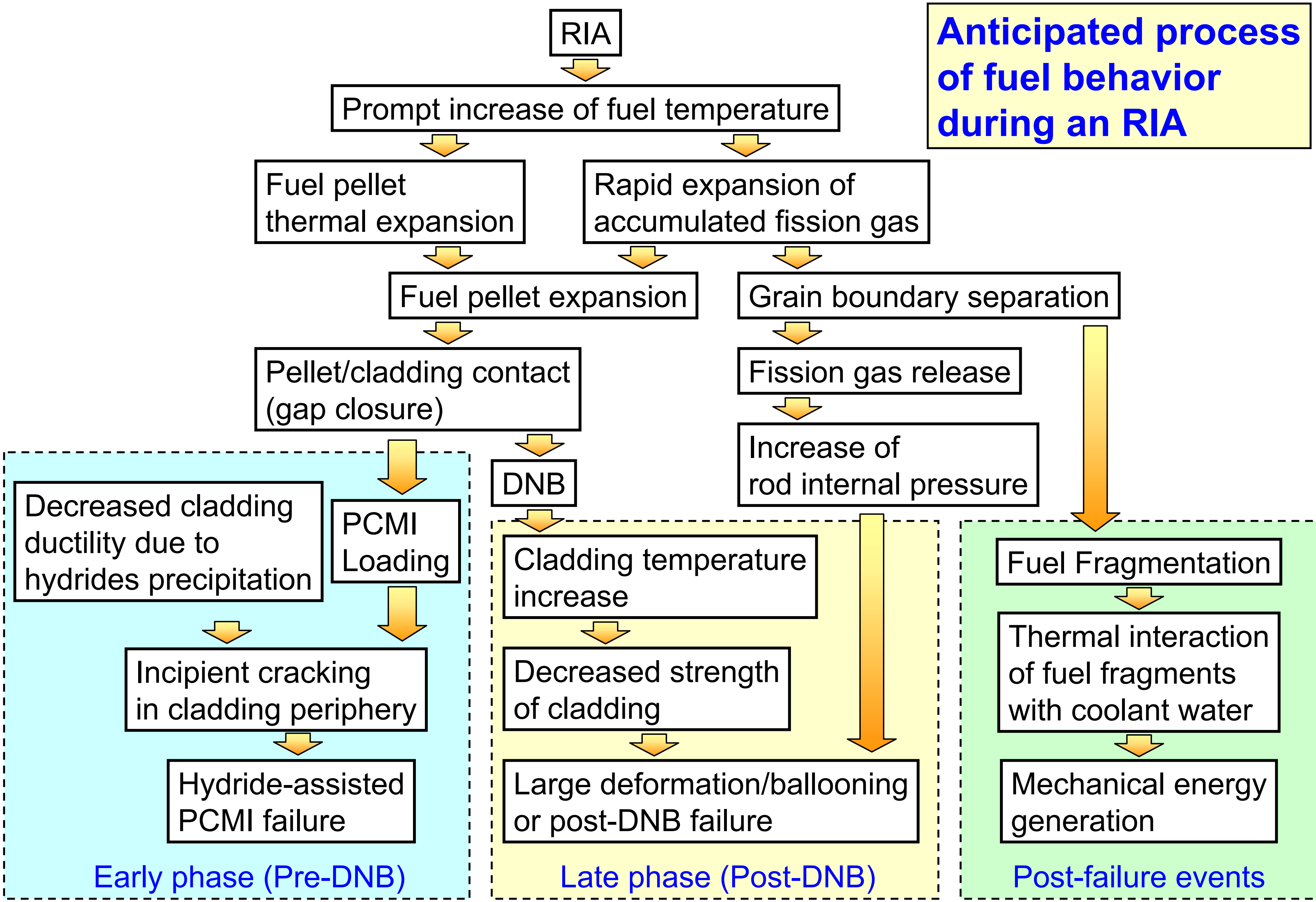
# NSRR Experiments with irradiated LWR fuels

| Test fuels | Fuel burnup (MWd/kg) 10 20 30 40 50 60 | Number of tests |
|---|---|---|
| PWR (14x14, 17x17) | | 26 |
| BWR (7x7, 8x8) | | 16 |
| ATR/MOX | | 5 |
| JMTR pre-irradiated | | 22 |





- ✓ **Hydride-assisted PCMI failure**
- ✓ **Fuel dispersal and mechanical energy generation**
- ✓ **Large rod expansion and fission gas release**
- ✓ **Possible MOX effect (Role of Plutonium agglomerates)**

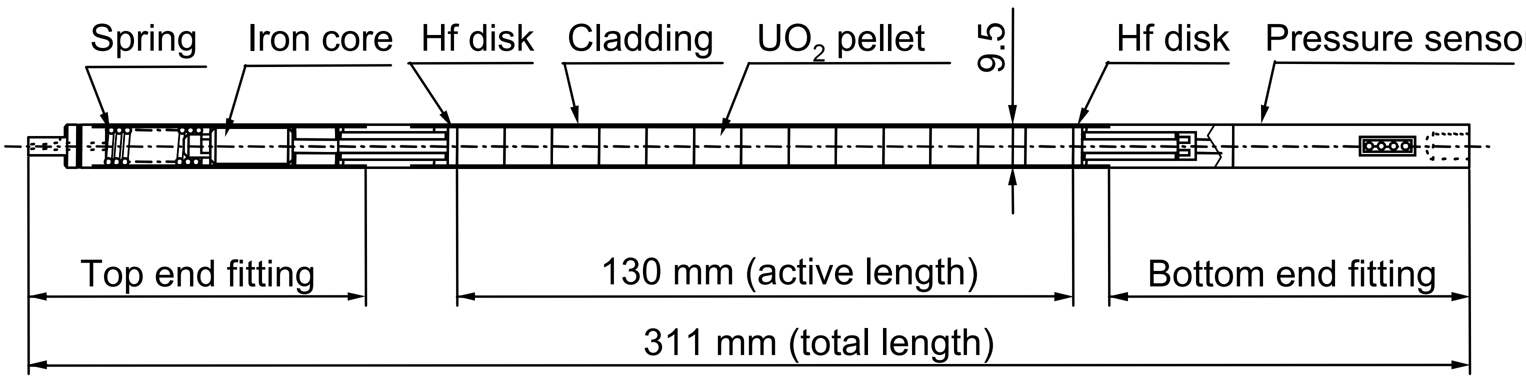

Anticipated process of fuel behavior during an RIA
RIA
Prompt increase of fuel temperature
Fuel pellet thermal expansion
Rapid expansion of accumulated fission gas
Fuel pellet expansion
Grain boundary separation
Pellet/cladding contact (gap closure)
Fission gas release
Increase of rod internal pressure
DNB
Decreased cladding ductility due to hydrides precipitation
PCMI Loading
Incipient cracking in cladding periphery
Hydride-assisted PCMI failure
Early phase (Pre-DNB)
Cladding temperature increase
Decreased strength of cladding
Large deformation/ballooning or post-DNB failure
Late phase (Post-DNB)
Fuel Fragmentation
Thermal interaction of fuel fragments with coolant water
Mechanical energy generation
Post-failure events

# Tests OI-10 and OI-11



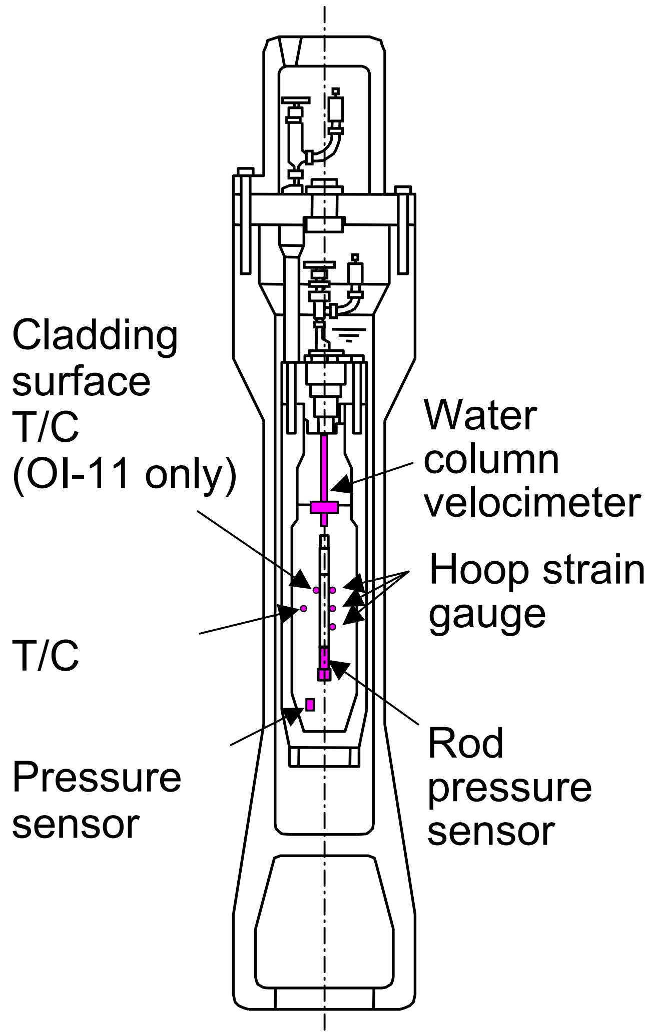
PWR 55 GWd/t lead-use fuel rods (Ohi unit 4)

| | | |
|---|---|---|
| Test ID | OI-10 | OI-11 |
| Fuel type | PWR 17x17 | |
| Cladding material | MDA | ZIRLO |
| Initial enrichment | 4.5% | 4.5% |
| Pellet grain size (μm) | ~25 | ~8 |
| Operation period | 4 cycles from Mar.97 to Mar.02 | |
| Test rod sampling position | 2nd span from the top | |
| Test rod burnup (GWd/t) | 60 | 58 |
| Average / Max. heat rates (kW/m) | 15.6 / 19.5 | 15.2 / 20.3 |
| Heat rate in last cycle (kW/m) | 13.0 | 13.2 |
| Cladding oxide thickness (μm) | ~30 | ~30 |



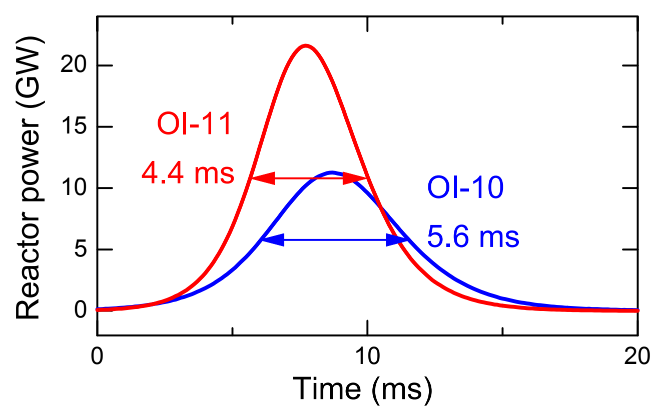

Rod of OI-10 is 10 mm shorter in total and active lengths.

# Tests OI-10 and OI-11




Instrumentation

| | OI-10 | OI-11 |
|---|---|---|
| Coolant conditions | Stagnant ~20 deg C, 0.1 MPa | |
| Pulse irradiation | | |
| Inserted reactivity($) | 3.67 | 4.6 |
| Peak fuel enthalpy (J/g) | 435 | 657 |
| (cal/g) | 104 | 157 |
| Test results | No failure<br>No significant deformation | Failure at 120 cal/g<br>Cladding axial crack<br>Fuel fragmentation |



# Test OI-11

PWR

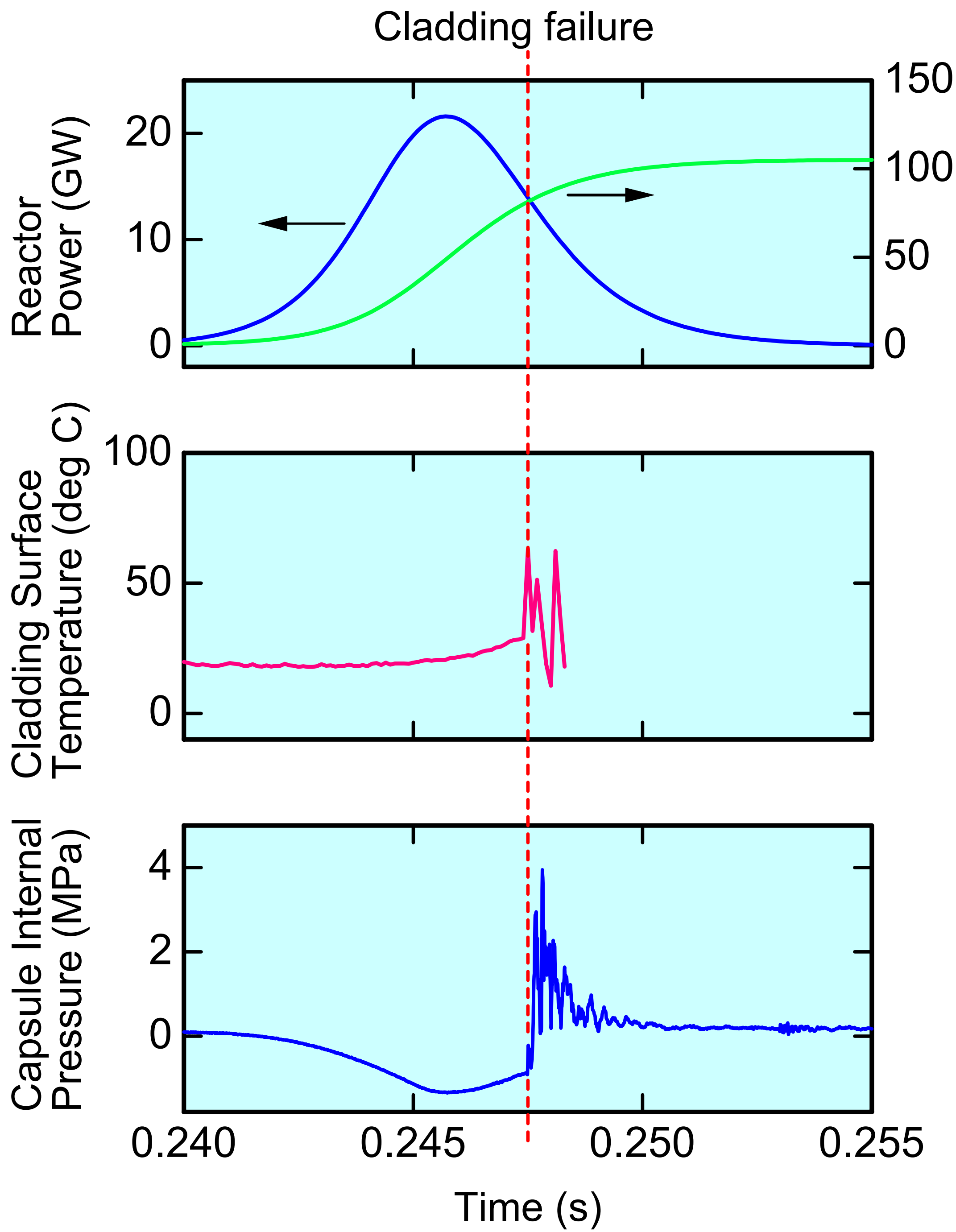
58 GWd/tU

$P_{rod,\ ini}$ 0.1 MPa

Peak Enthalpy
157 cal/g

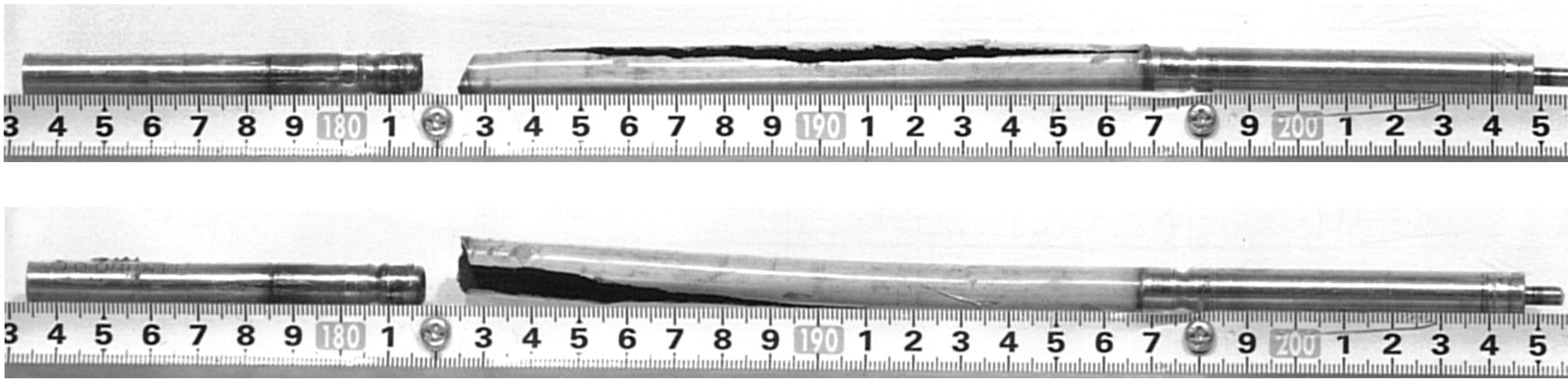
Failed at 120 cal/g

7



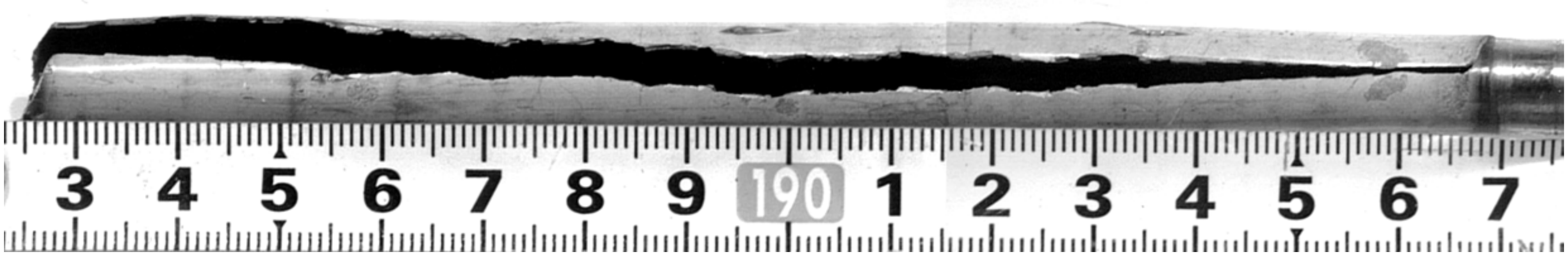
# Test OI-11
## - Post-pulse rod appearance -

Bottom Top



Fracture close to the welding position

Axial crack over the fuel stack



# Tests OI-10 and OI-11 - Summary 1/2 -



- ✓ High burnup PWR fuels with new cladding were subjected to the NSRR experiments. Test OI-10 rod has an MDA (Mitsubishi Developed Alloy, Zr-0.8Sn-0.2Fe-0.1Cr-0.5Nb) cladding and Test OI-11 rod has a ZIRLO™ cladding.

- ✓ A test rod of the OI-10 has a burnup of 60 GWd/t and cladding oxide thickness of ~30 μm. The fuel was pulse-irradiated with conditions of 104 cal/g (0.44 kJ/g) for a peak fuel enthalpy and 5.6 ms for a pulse-width. The fuel remained intact in the OI-10.

# Tests OI-10 and OI-11
# - Summary 2/2 -

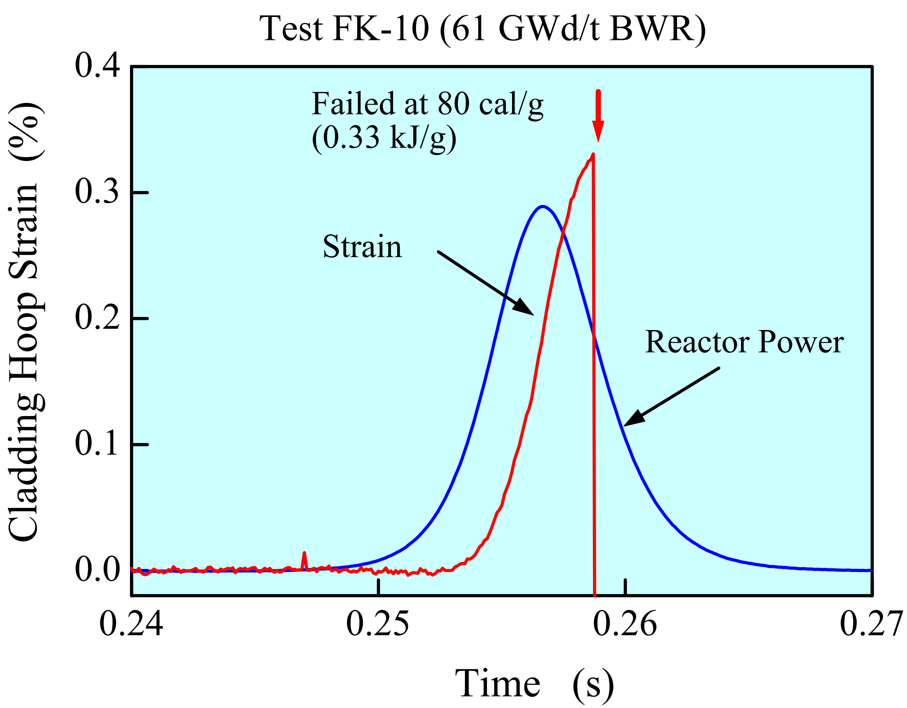


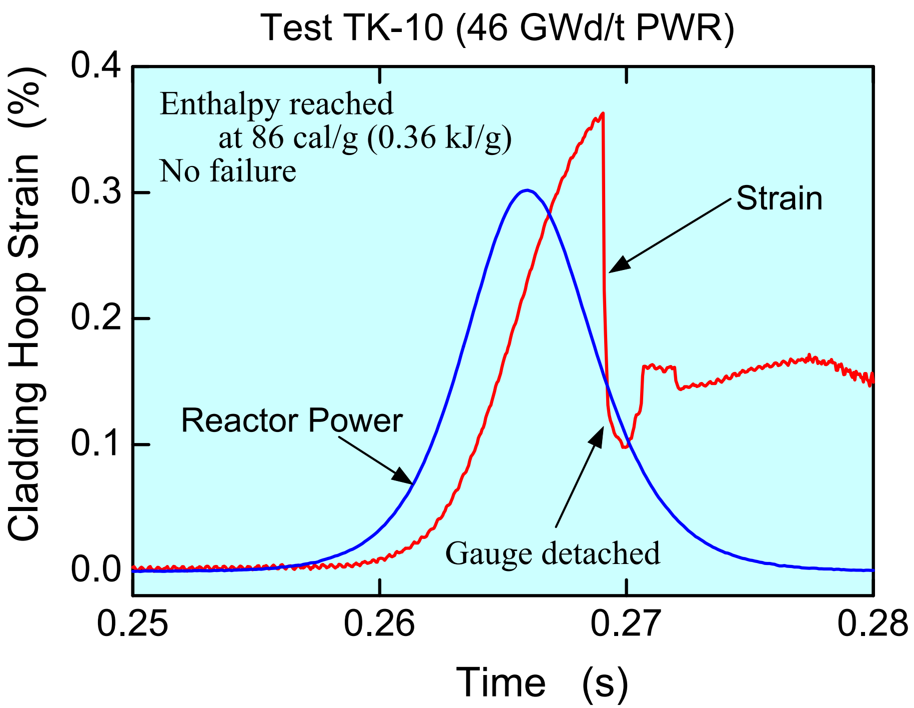
- ✓ A test rod of the subsequent OI-11 has a burnup of 58 GWd/t and cladding oxide thickness of ~30 μm. The fuel was tested with conditions of 157 cal/g (0.66 kJ/g) for a peak fuel enthalpy and 4.4 ms for a pulse-width. The Test OI-11 resulted in fuel failure, pellets fragmentation and mechanical energy generation. Transient records showed that a fuel enthalpy at a time of failure was higher than those observed in previously tested fuels with Zircaloy-4 cladding and exceeded 120 cal/g (0.50 kJ/g).

- ✓ Results from the two tests, no failure in the OI-10 and the higher failure energy in the OI-11, reflects the better performance of these new cladding materials in terms of corrosion, the thinner oxides and accordingly lower hydrogen content generated during irradiation in the PWR.

# Transient Hoop Strain Measurement

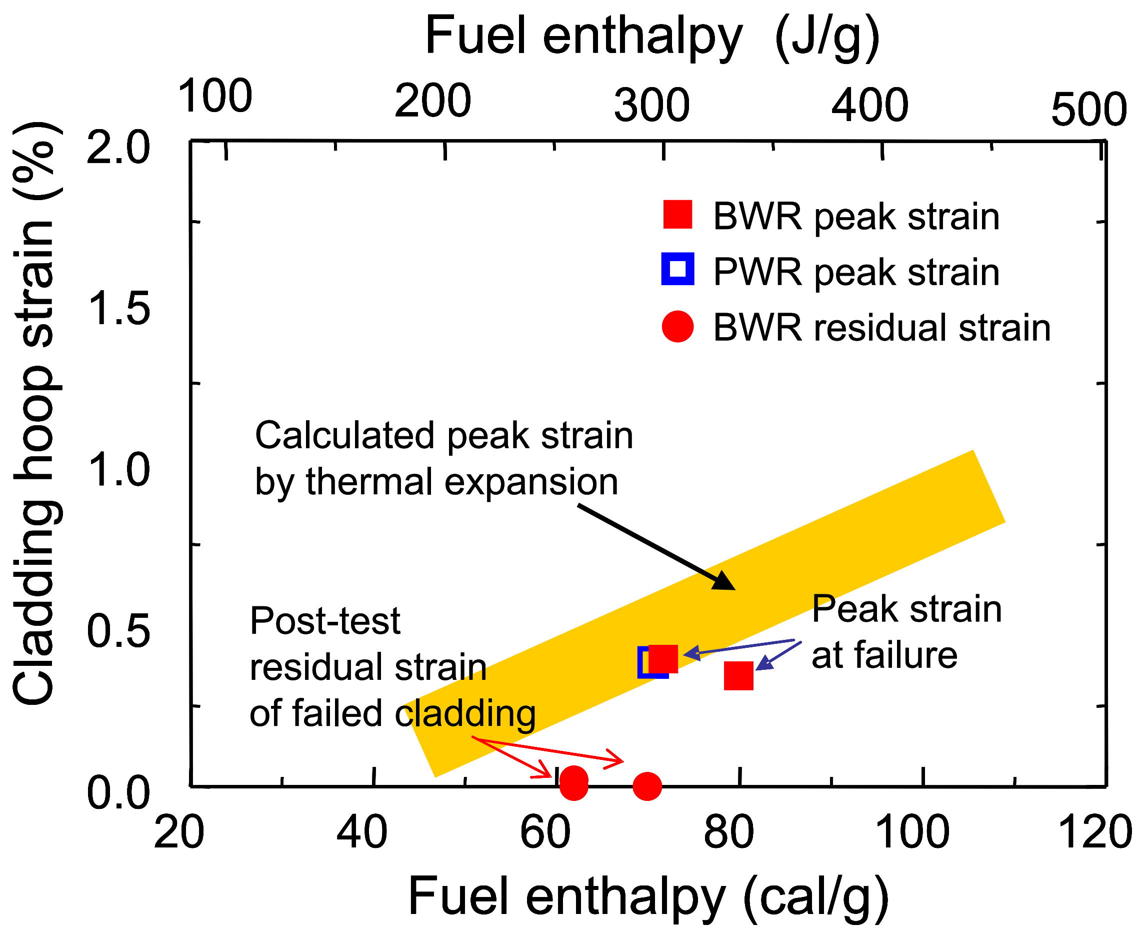
- ✓ Transient hoop deformation due to PCMI in early phase of RIA transient was measured with strain gauges on irradiated fuel rod.
- ✓ The hoop stain was about 0.4% at fuel enthalpy of about 80cal/g.



# Cladding strains at failure

- Peak strain measured in 70 to 80 cal/g was below 0.4%.
- Residual strain of failed cladding was ~0%.




The deformation resulting in cladding failure in early phase of transient can be explained only by thermal expansion of fuel pellets

# Tests with artificially-hydrided cladding



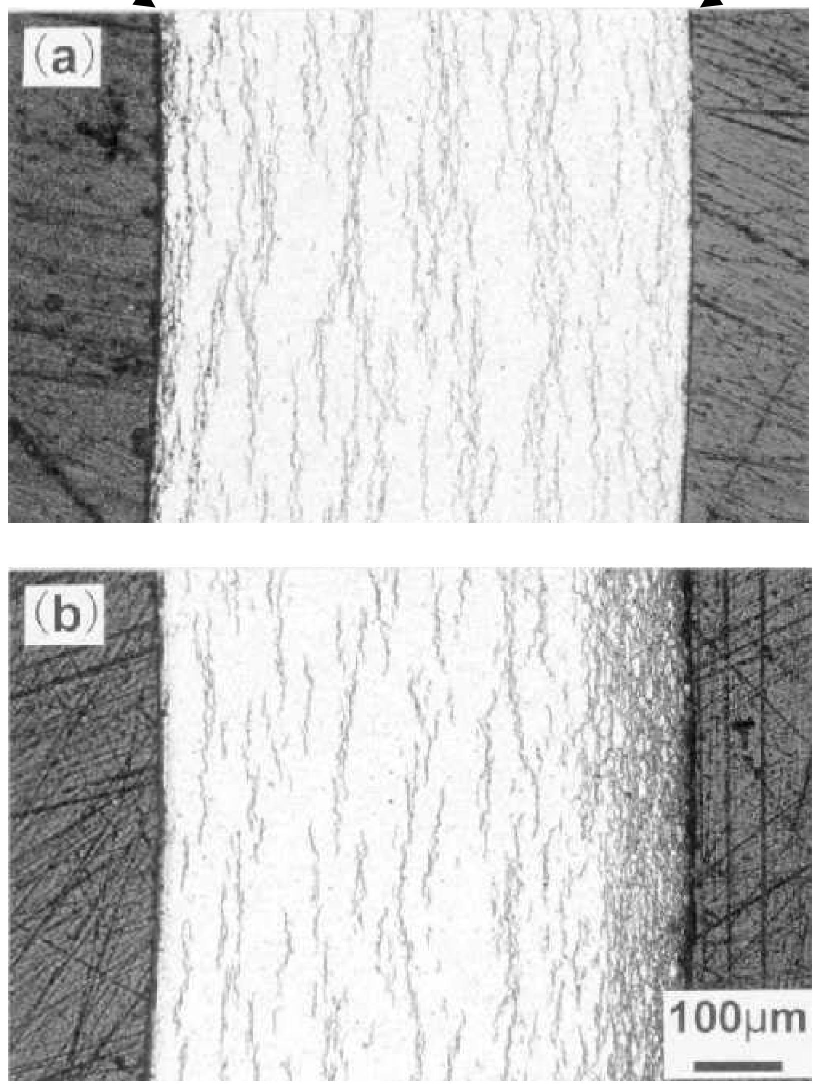
## Radial cross-section of artificially hydrided cladding samples

Inner surface

Outer surface

**Uniformly Hydrided Sample**

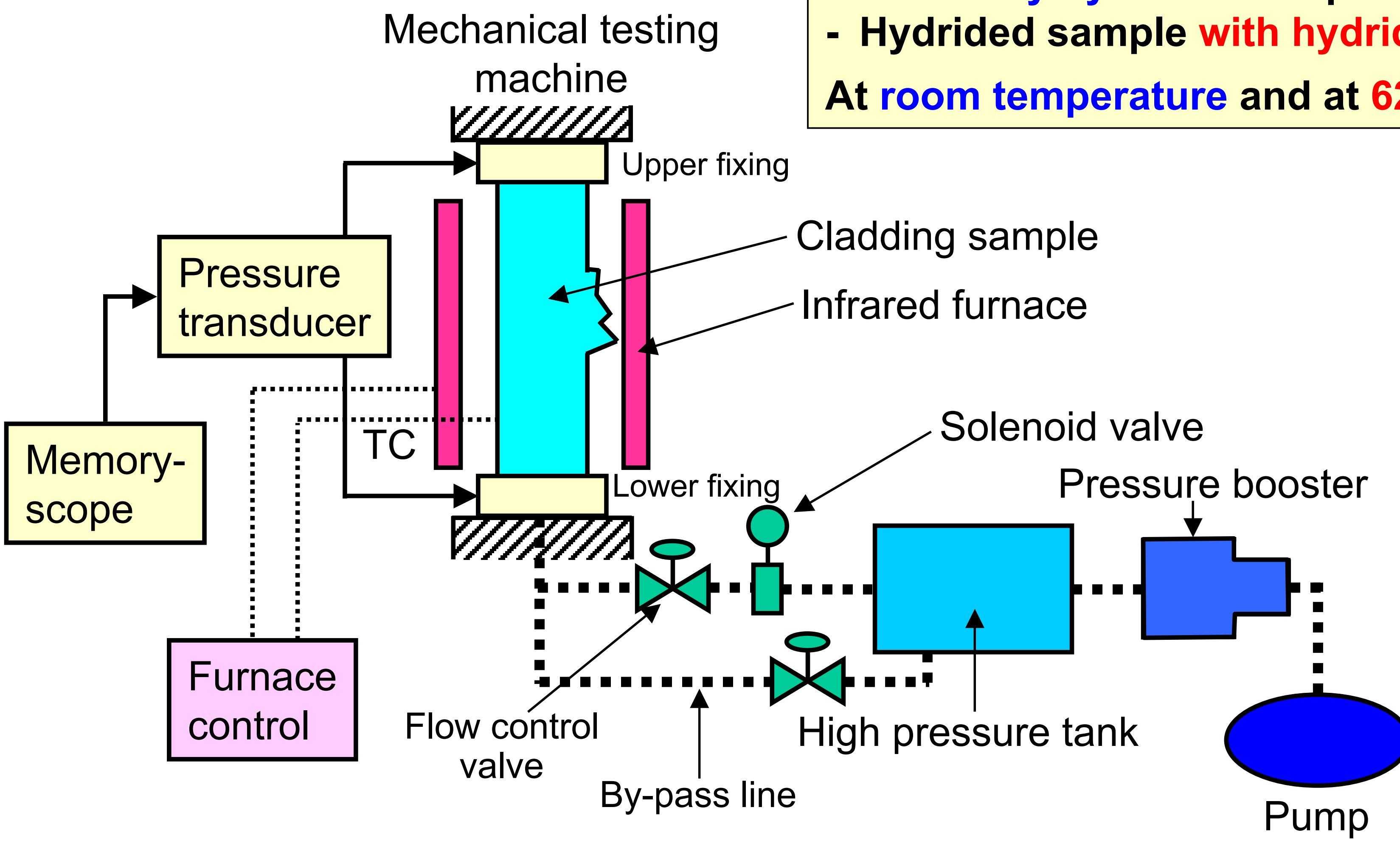
**Sample with Hydride Rim**



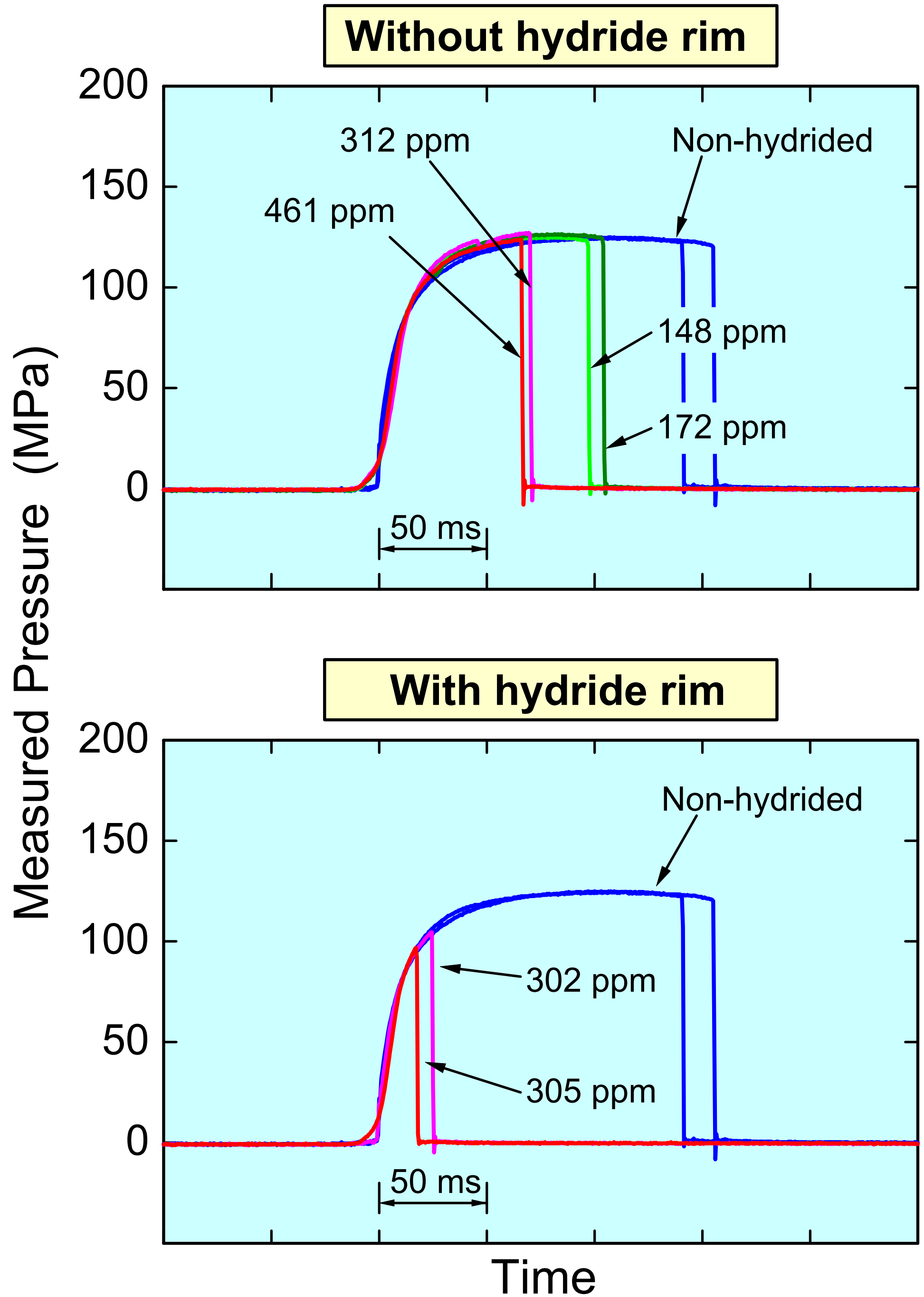
# Tube Burst Test

**Un-irradiated 17x17 PWR low tin Zry-4 cladding**
- **As-received, non-hydrided sample**
- **Uniformly hydrided sample**
- **Hydrided sample with hydride rim**

**At room temperature and at 620 K**



## Transient histories of sample internal pressure during tube burst tests

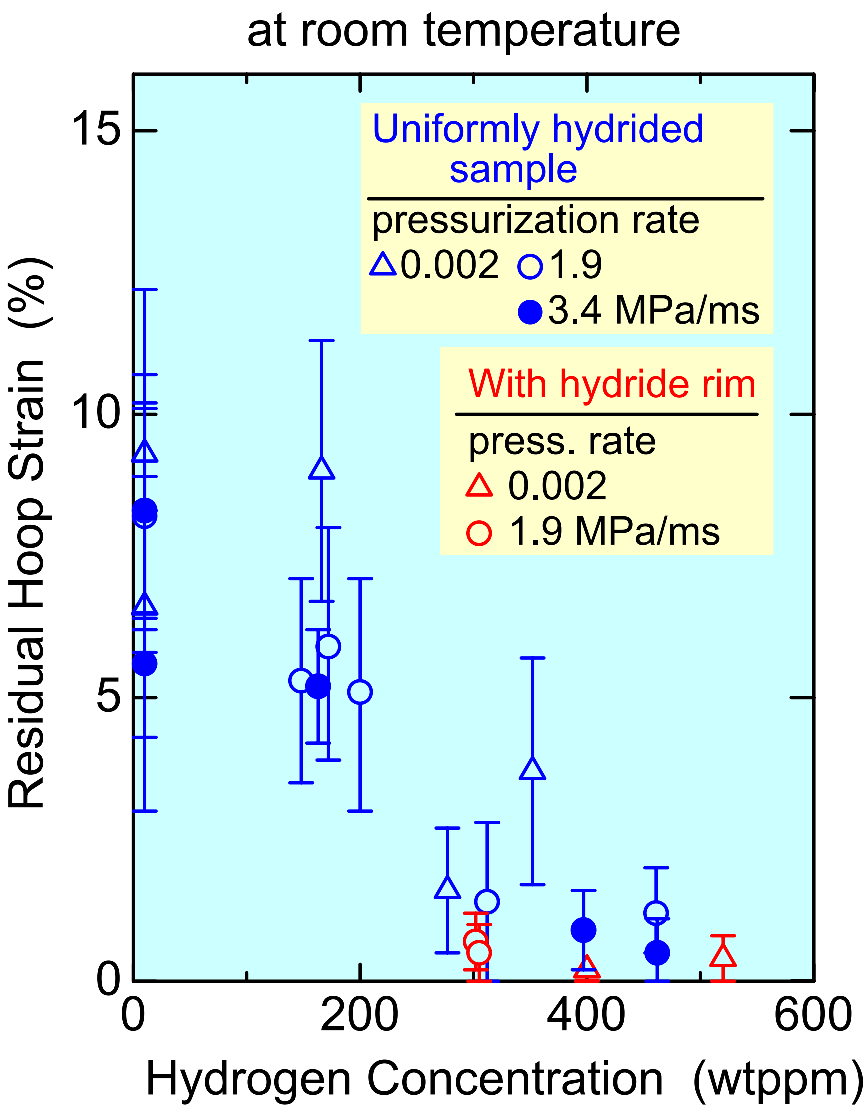


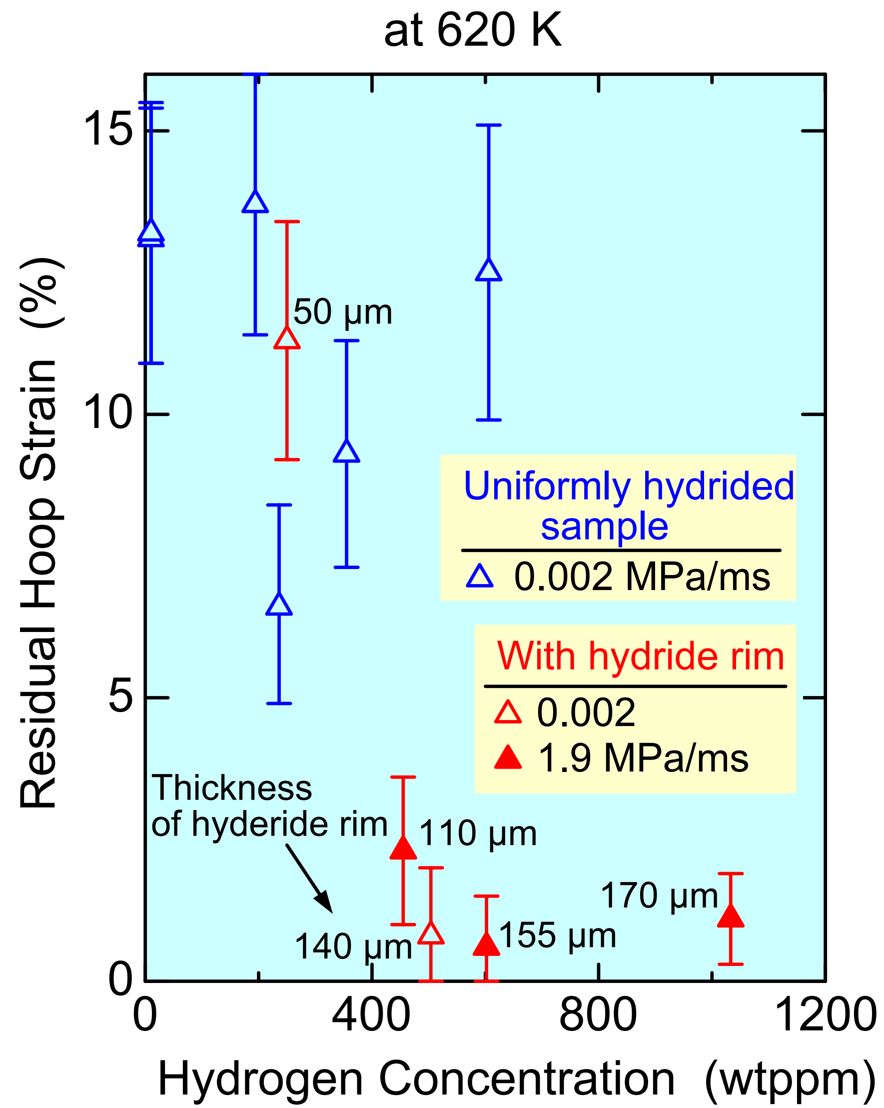

1.9 MPa/ms
room temperature

Tube Burst Test

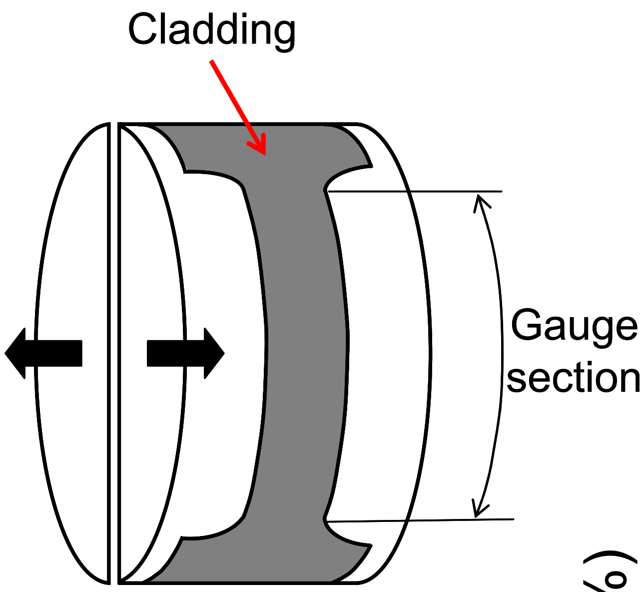


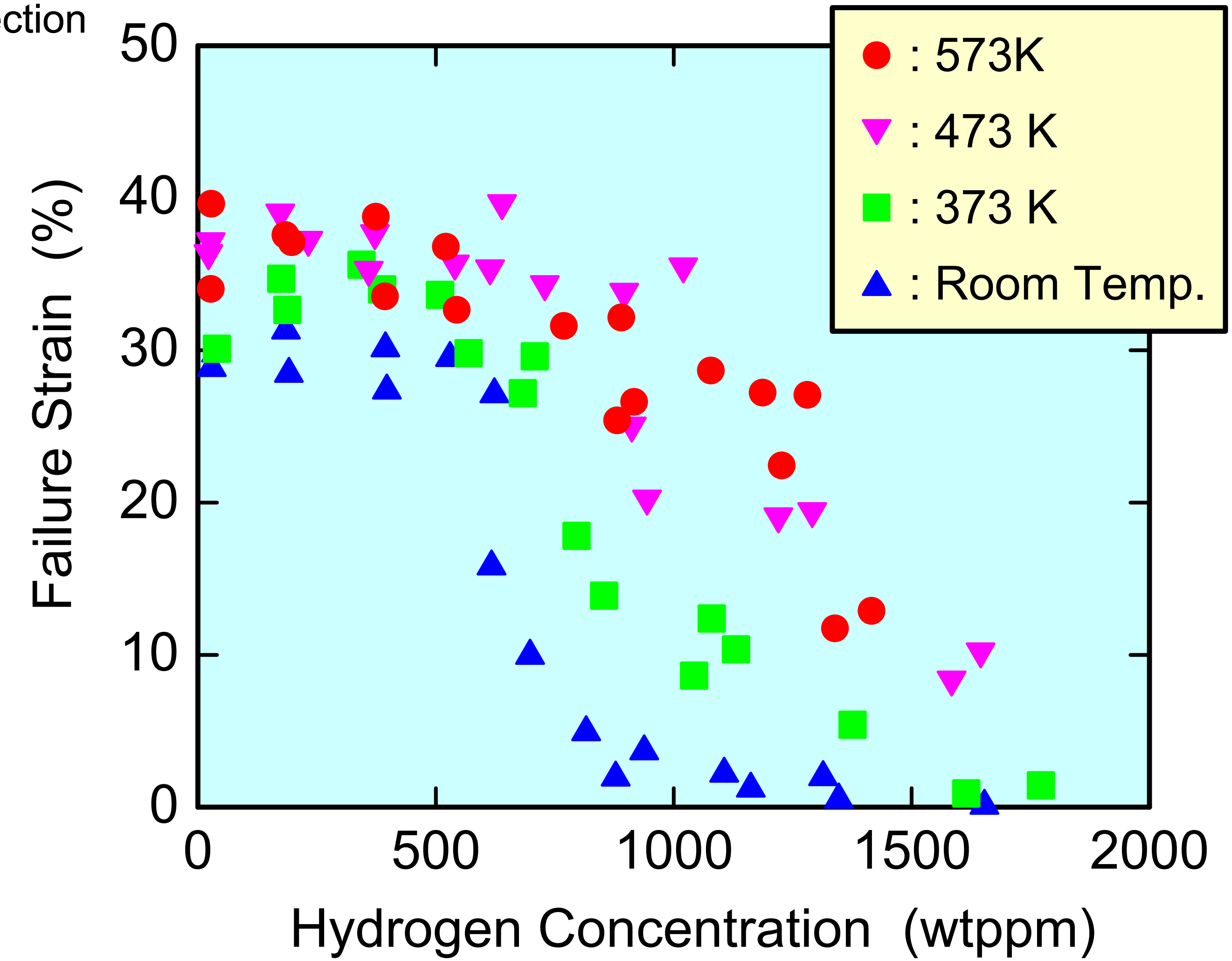
# Residual Hoop Strain
## as a function of
# Hydrogen Concentration



# Ring Tensile Test



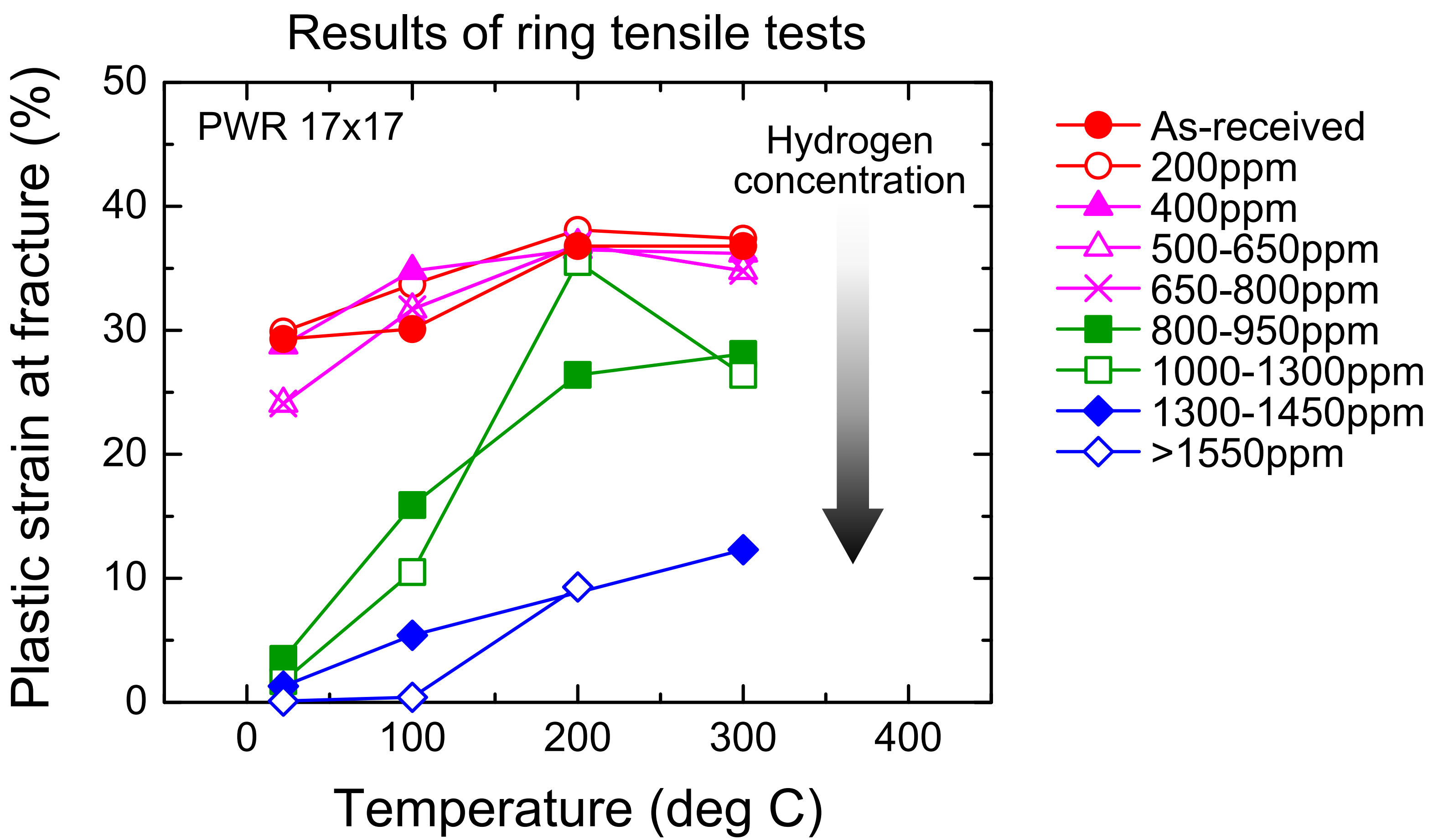
# Fuel rods to be tested in FY2004 to 2007

| Specification | | | Power plant (country) | Burnup* | Cladding | Number of RIA test | |
|---|---|---|---|---|---|---|---|
| Fuel | Reactor | Type | | GWd/t | | RTRP** | HTHP*** |
| $UO_2$ | PWR | 17×17 | OI, unit 4 (Japan) | ~60 | NDA | 1 | 0 |
| | | | Vandellos (Spain) | 74 | MDA | 2 | 1 |
| | | | | | ZIRLO | 1 | 0 |
| | | | McGuire (USA) R2 (Sweden) | 71 | NDA | 1 | 0 |
| | | | Graveline (France) | 66 - 69 | M5 | 1 | 1 |
| | BWR | 10×10 | Leibstadt (Switzerland) | 63 | Zry-2 | 1 | 1 |
| MOX | ATR | - | Fugen (Japan) | 43 | Zry-2 | 1 | 0 |
| | PWR | 14×14 | Beznau (Switzerland) | 59 | Zry-4 | 1 | 1 |
| | | | | 44 | Zry-4 | 1 | 0 |
| | BWR | 8×8 | Dodewaard (Netherlands) | 46 | Zry-2 | 1 | 0 |
| | | | | | | 11 | 4 |

* Segment average for OI, Fugen and R2, rod average for the others.

** Room-temperature/pressure. *** High-temperature/pressure.

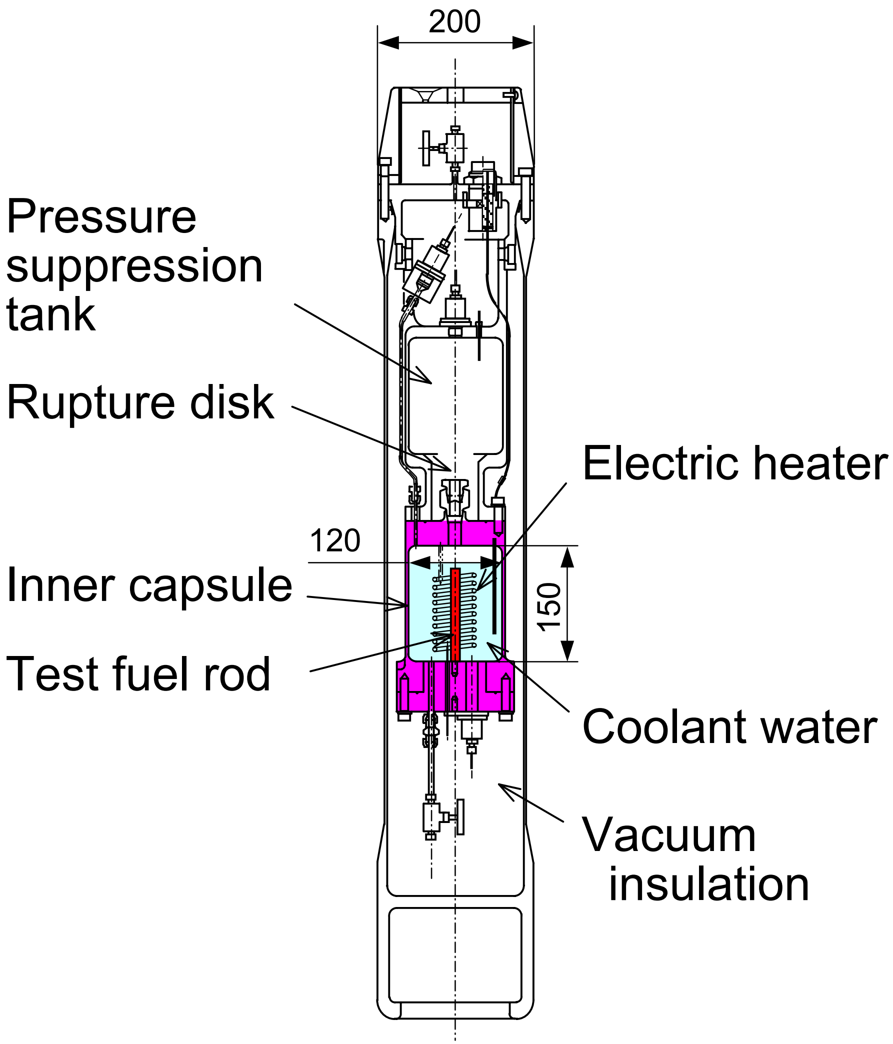
# Temperature effect on cladding ductility




- Influence of hydrogen concentration and temperature on the cladding ductility

➢ High-temperature capsule in NSRR experiments

# High temperature capsule




HTHP capsule

Test fuel rod

- Rod length ~120 mm
- Stack length ~50 mm

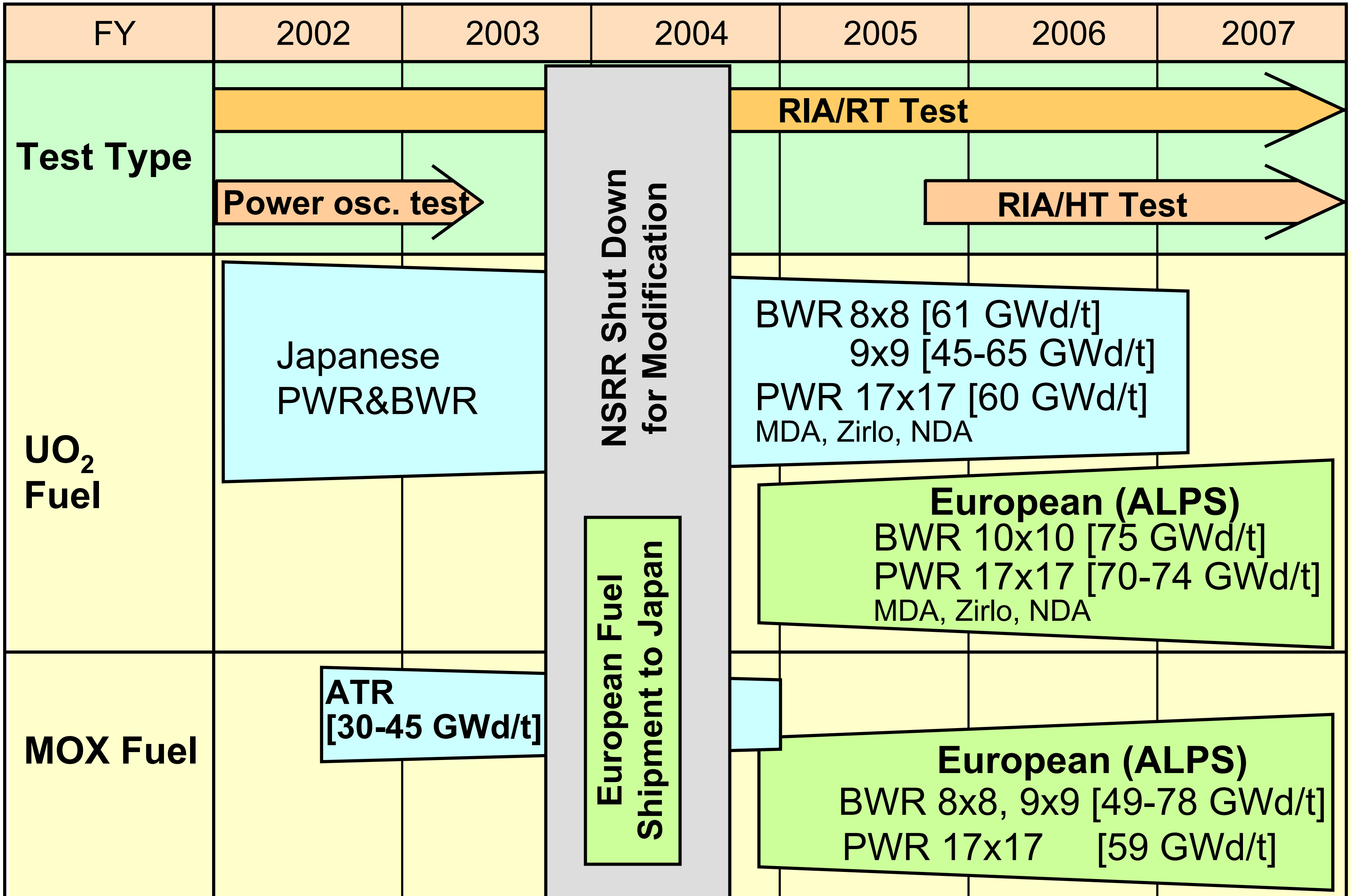
Coolant water

- Stagnant
- 286 deg C, 7 MPa (BWR conditions)

Instrumentation

- Cladding surface thermocouple
- Coolant thermocouple
- Capsule pressure sensor

# NSRR Test Schedule



# NSRR experiments

## Data at higher burnups

(pellet burnup: GWd/t)

| | | FY2003 | | |
|---|---|---|---|---|
| $UO_2$ : | PWR | 60 | ~80 | until FY2005 |
| | BWR | 61 | 66 | |
| MOX : | ATR | 30 | PWR 62 | until FY2006 |
| | | | BWR 48 | |

## Test at higher temperature

Coolant water temperature

From 20 (room temp.) to 90 deg C → 286 deg C

# CABRI CIP0-1 PRELIMINARY TEST RESULTS

JC Mélis, C Marquié, M Faury, J Papin

IRSN

CIP

IRSN

# THE CONTEXT

AEN NEA

➢Most utilities request, for economic reasons, an increase of the fuel discharge burn-up.

➢MOX fuel is largely introduced

➢Key points :

+ improve clad alloys  (M5, Zirlo, Duplex, MDA,…) which properties are limited by corrosion

+ improve MOX microstructure (TU2, SBR) to minimize fission gas release

CIP

# OBJECTIVES

- Determine $UO_2$ and MOX high burn-up fuel behaviour under RIA conditions
- Determine safety margins
- Propose new safety criteria more adapted to high burn up fuel

# THE CIP (1/2)

- In 2000, IRSN proposed a new program under the auspices of OECD
- 12 tests planned
- 14 organisations from 11 countries have signed an agreement with IRSN

CIP

# THE CIP (2/2)

- 2 first tests performed in the Na loop environment (reference)

  + CIP0-1  75 GWd/t cladded with Zirlo - 30 ms pulse width
  + CIP0-2 75 GWd/t cladded with M5 – 30 ms pulse width

- Both tests performed in November 2002

- CIP0-2 (M5) exhibited no clad failure

CIP

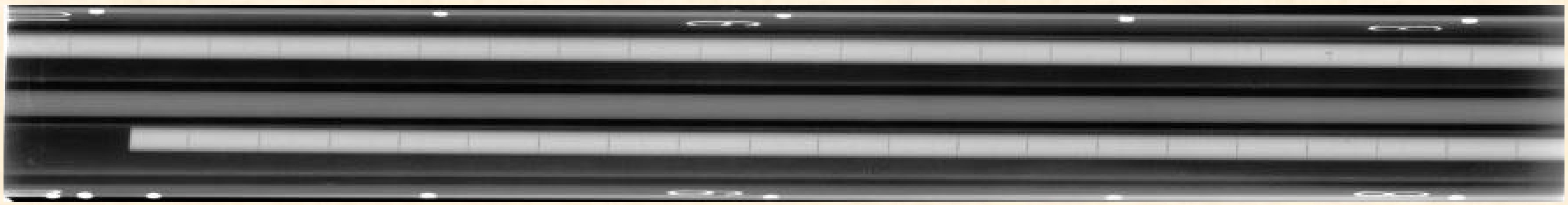
# CIP0-1 FUEL ROD

- Irradiated in the Vandellos reactor (Spain) up to 74.8 GWd/t (pellet burn-up)
- Examined and refabricated in the Studsvik Labs (Sweden)
- Shipped to Cadarache in June 2002

CIP

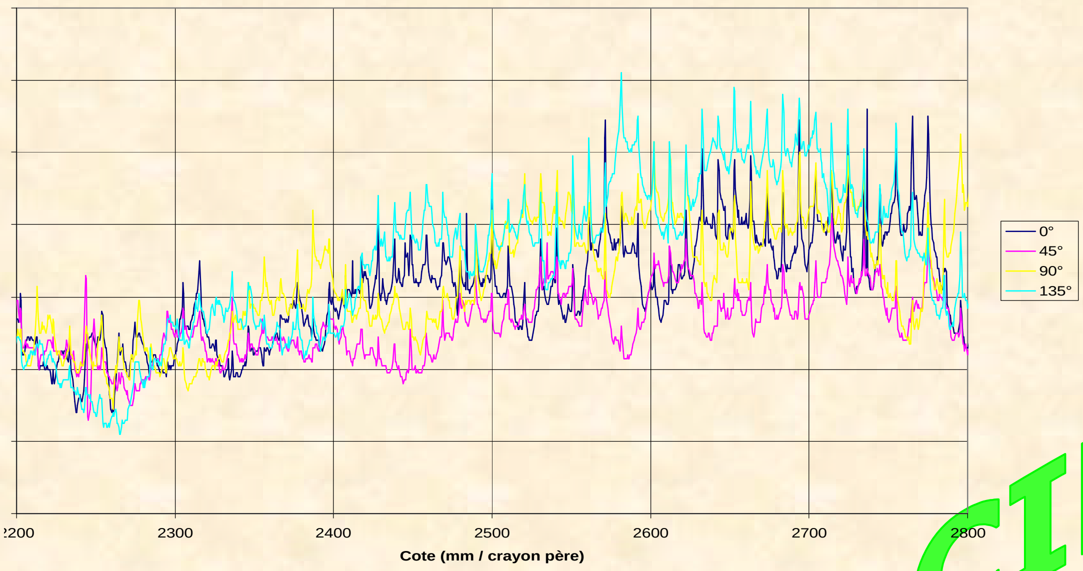
## Examinations

Neutronography : hydride concentrations at pellet-pellet interface



Metrology:

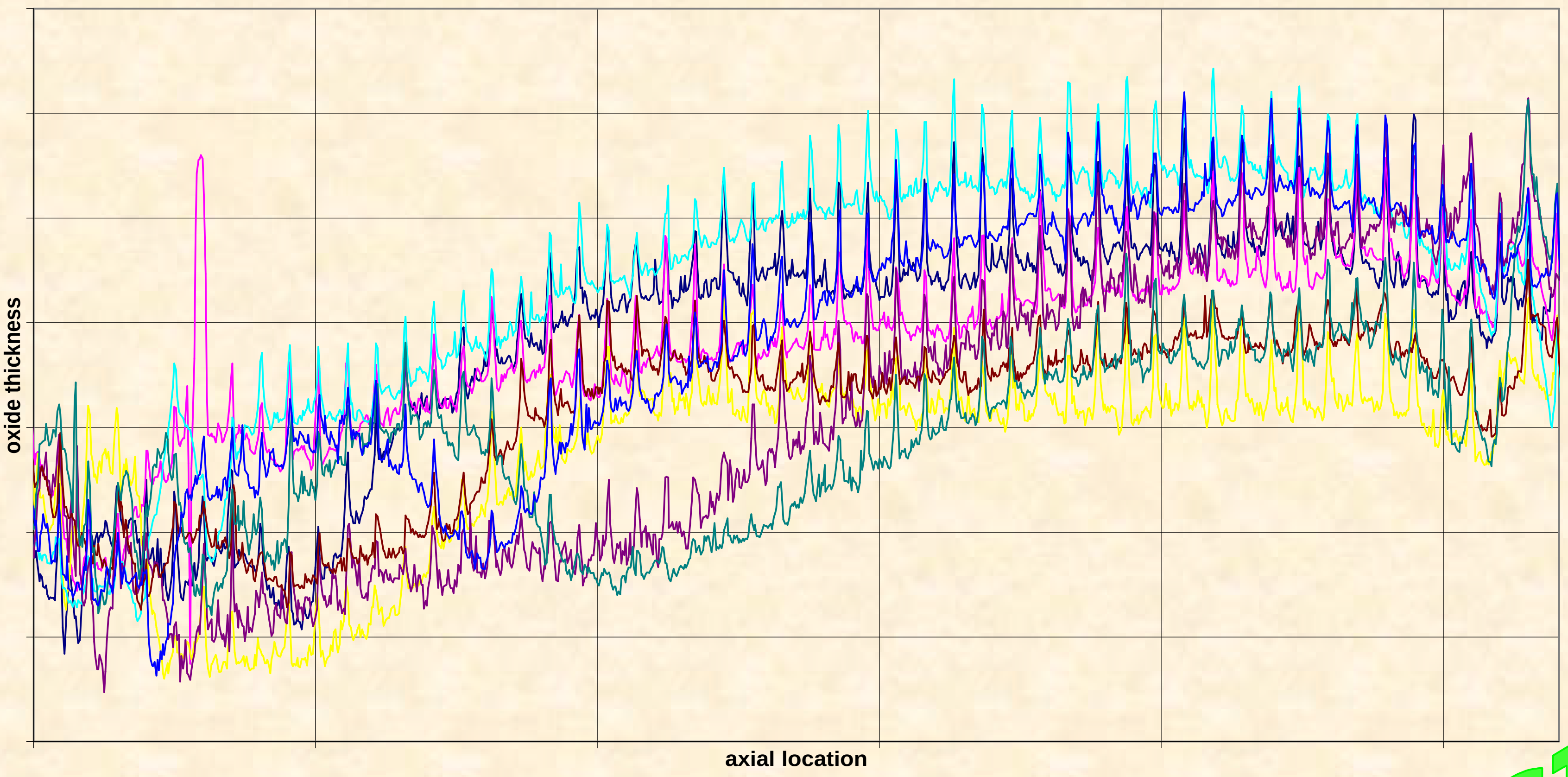
Typical of high burn-up fuel (ridges, ovalization)




CIP

# CIP0-1 Zirconia layer before test

Important zirconia layer (75 µm average) with large axial azimuthal variations



# CIP0-1 test conditions

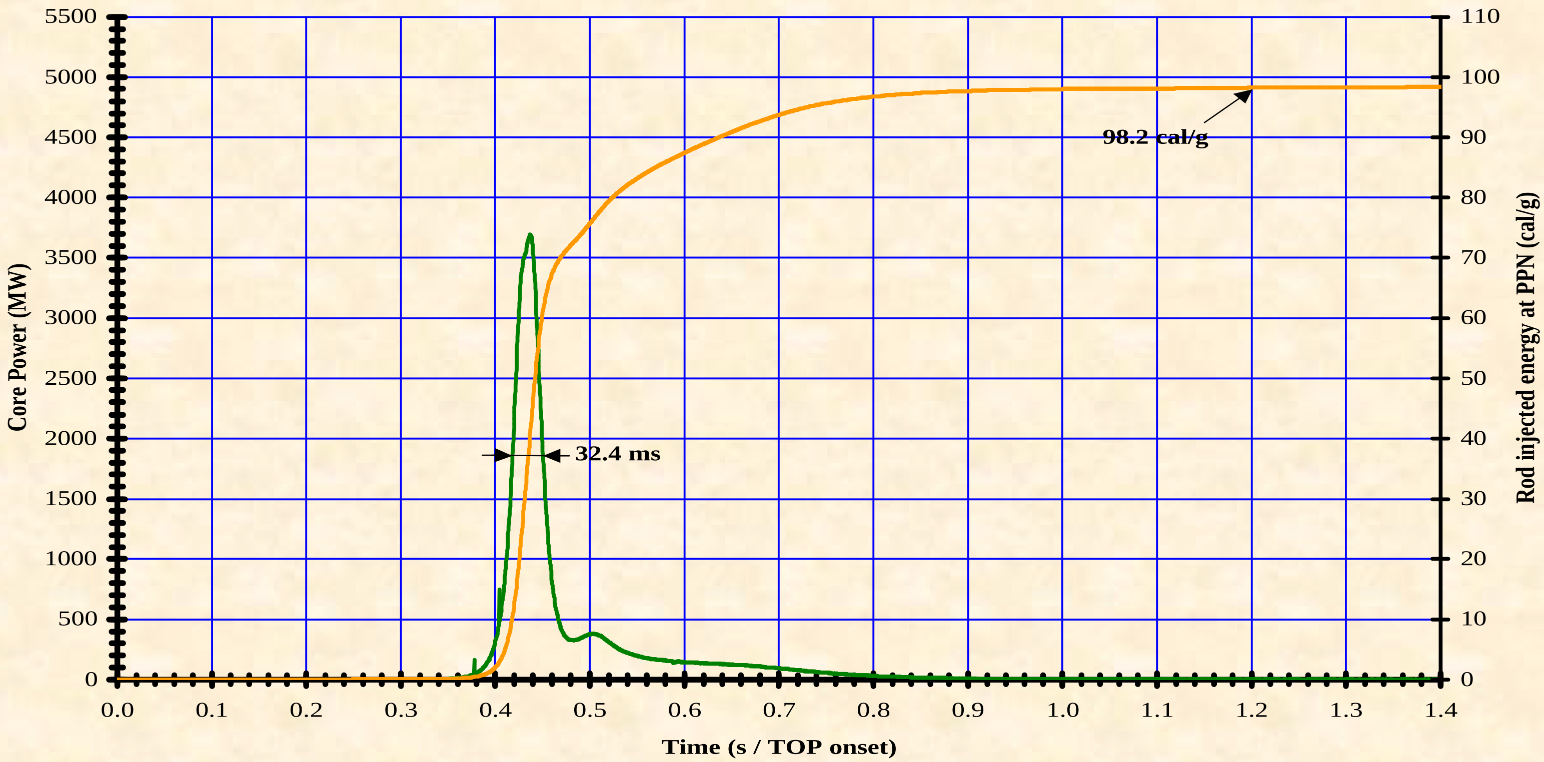
Mid-height width : 32.4 ms (30)

Total energy deposit : 98.2 cal/g (100)

Fluid velocity : 4 m/s

Coolant temperature : 280°C

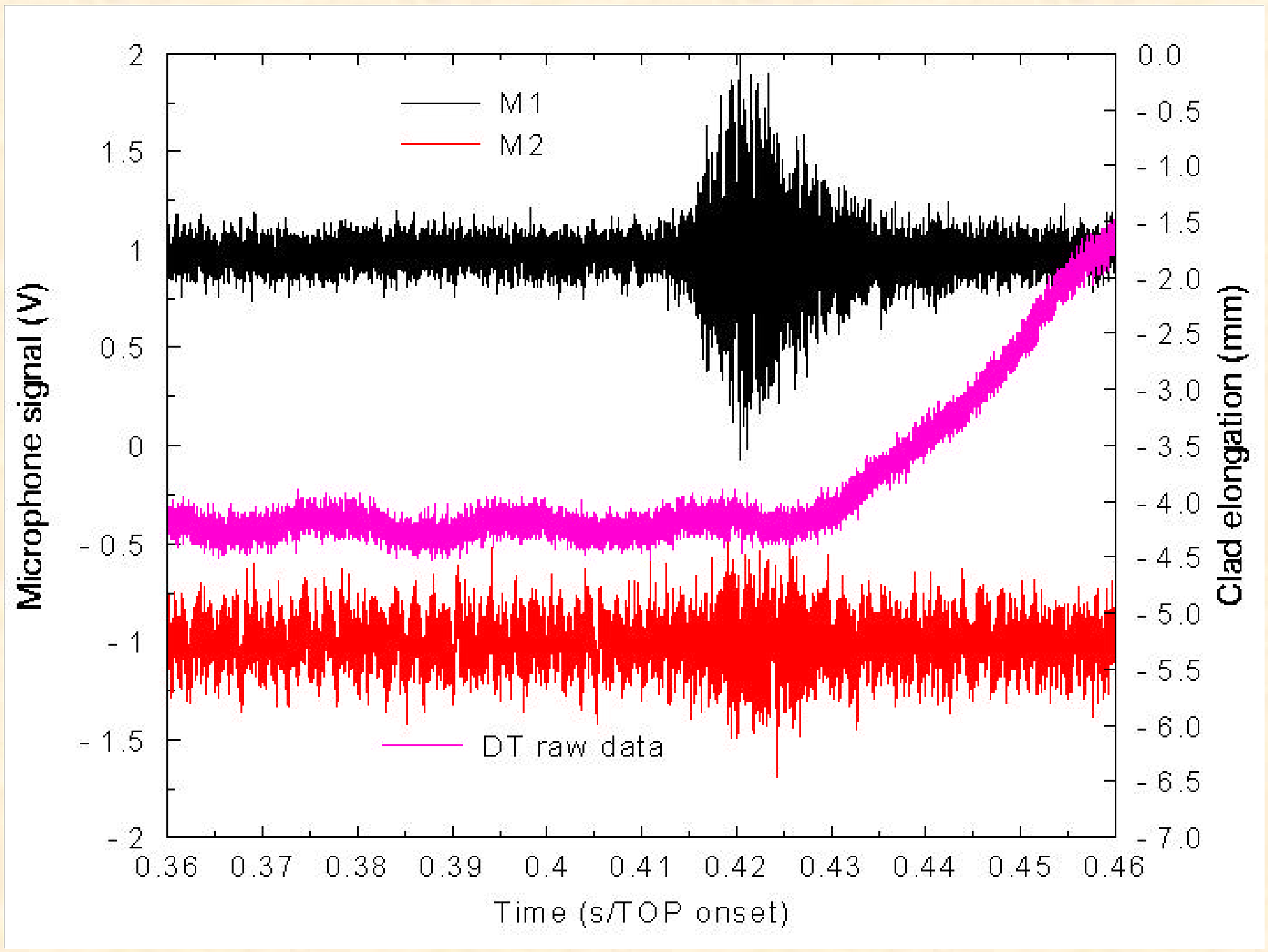
Core power : 100 kW

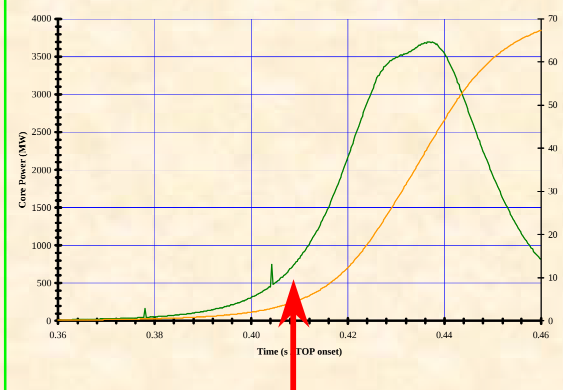



CIP

# CIP0-1 First event

## Microphone event linked with axial clad elongation




Time: TOP+0.41 s

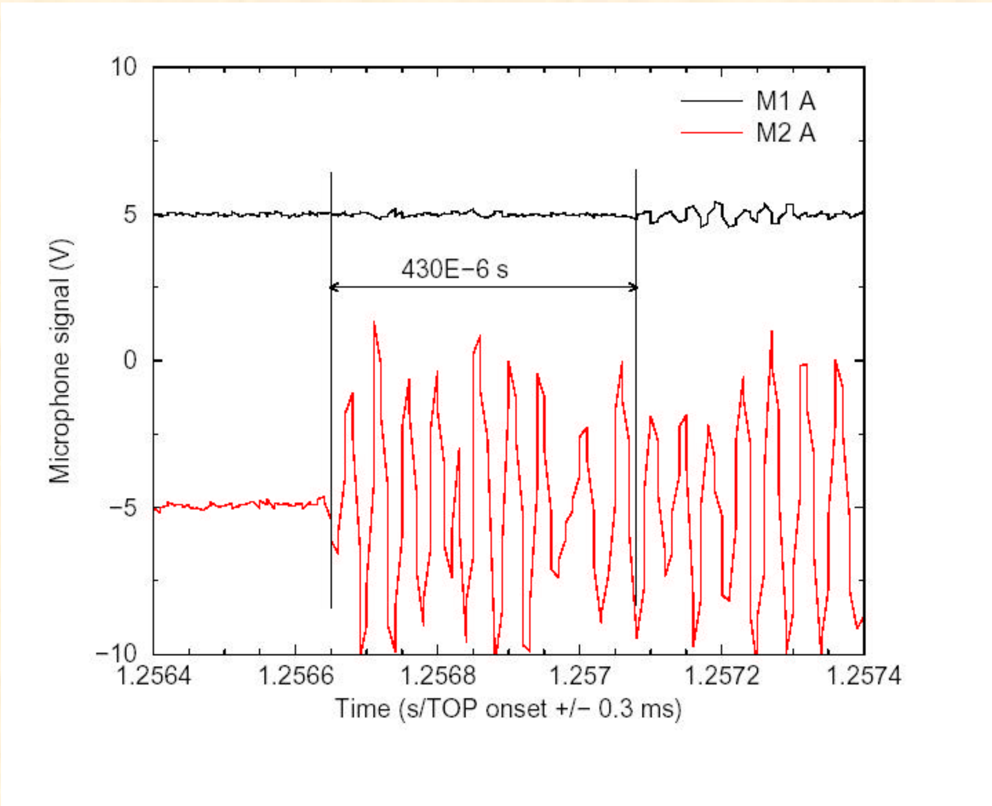
Energy: 8.5 cal/g

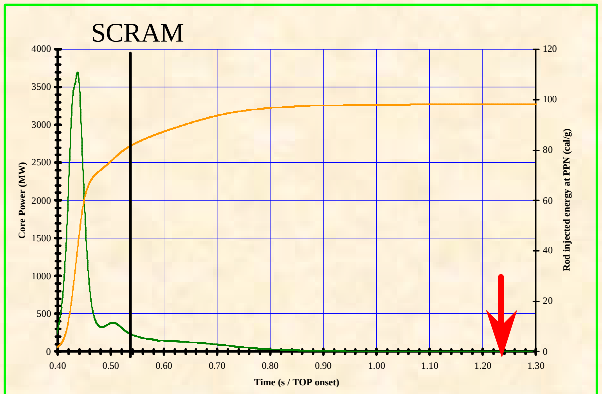
CIP

# CIP0-1 late event (1)



## Microphone event



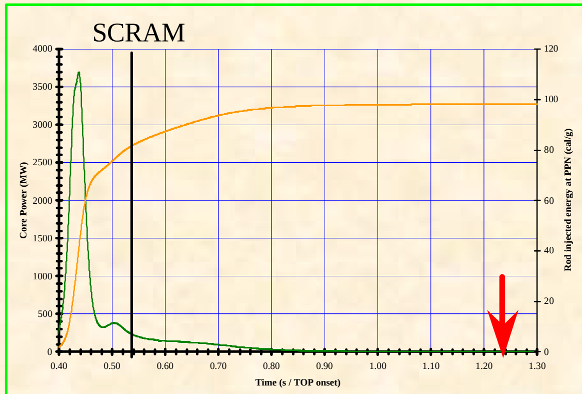

Time: TOP+1.26 s

Energy: 98.2 cal/g

Location well above the tested rod !

# CIP0-1 late event (2)

## Pressure event

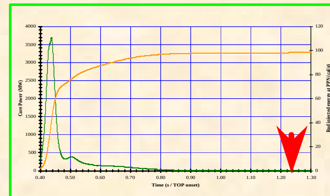




Time: TOP+1.26 s

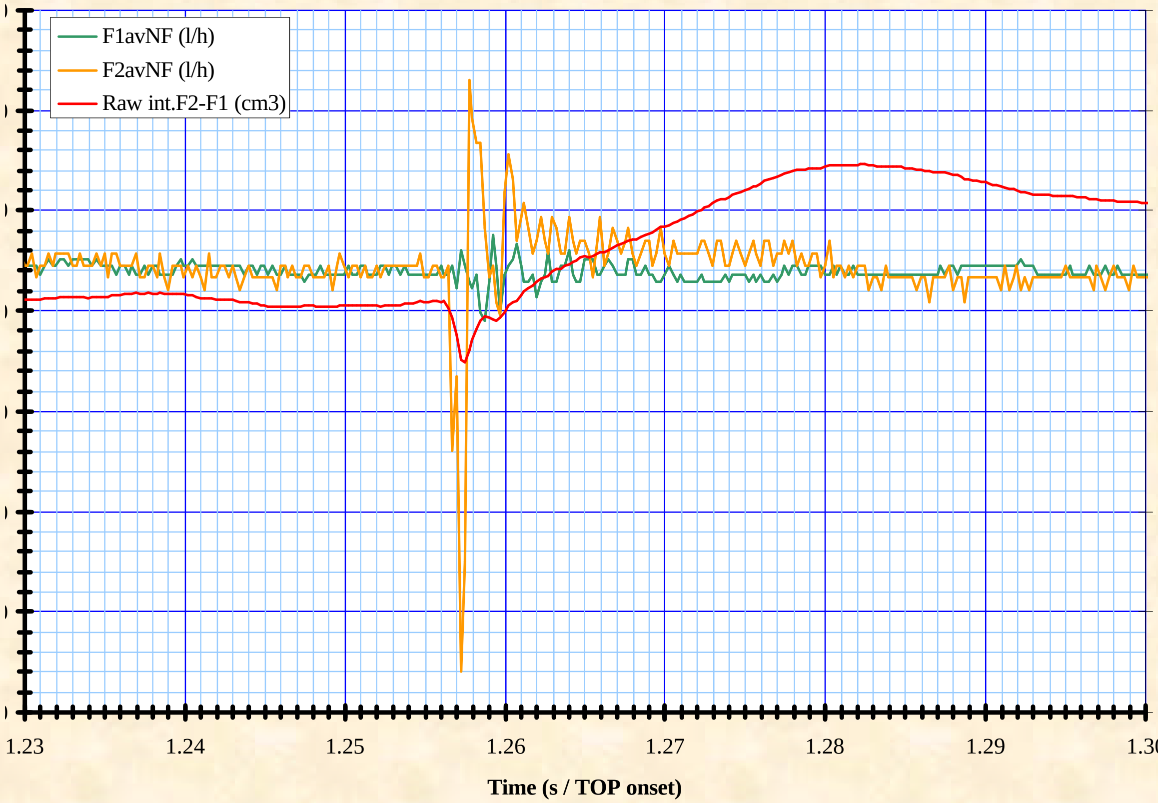
Energy: 98.2 cal/g

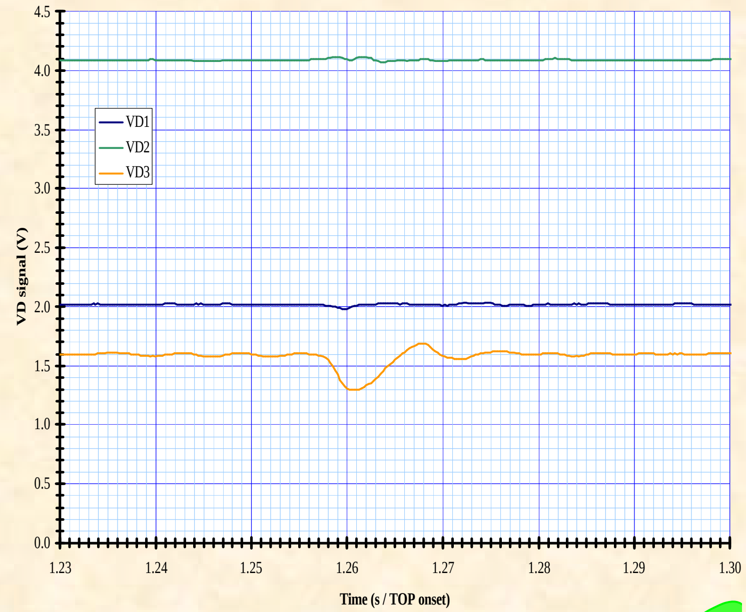
**150 µs before microphone event !!!**

# CIP0-1 late event (3)




## Flow and VD event




**150 µs before microphone event !!!**

RSN

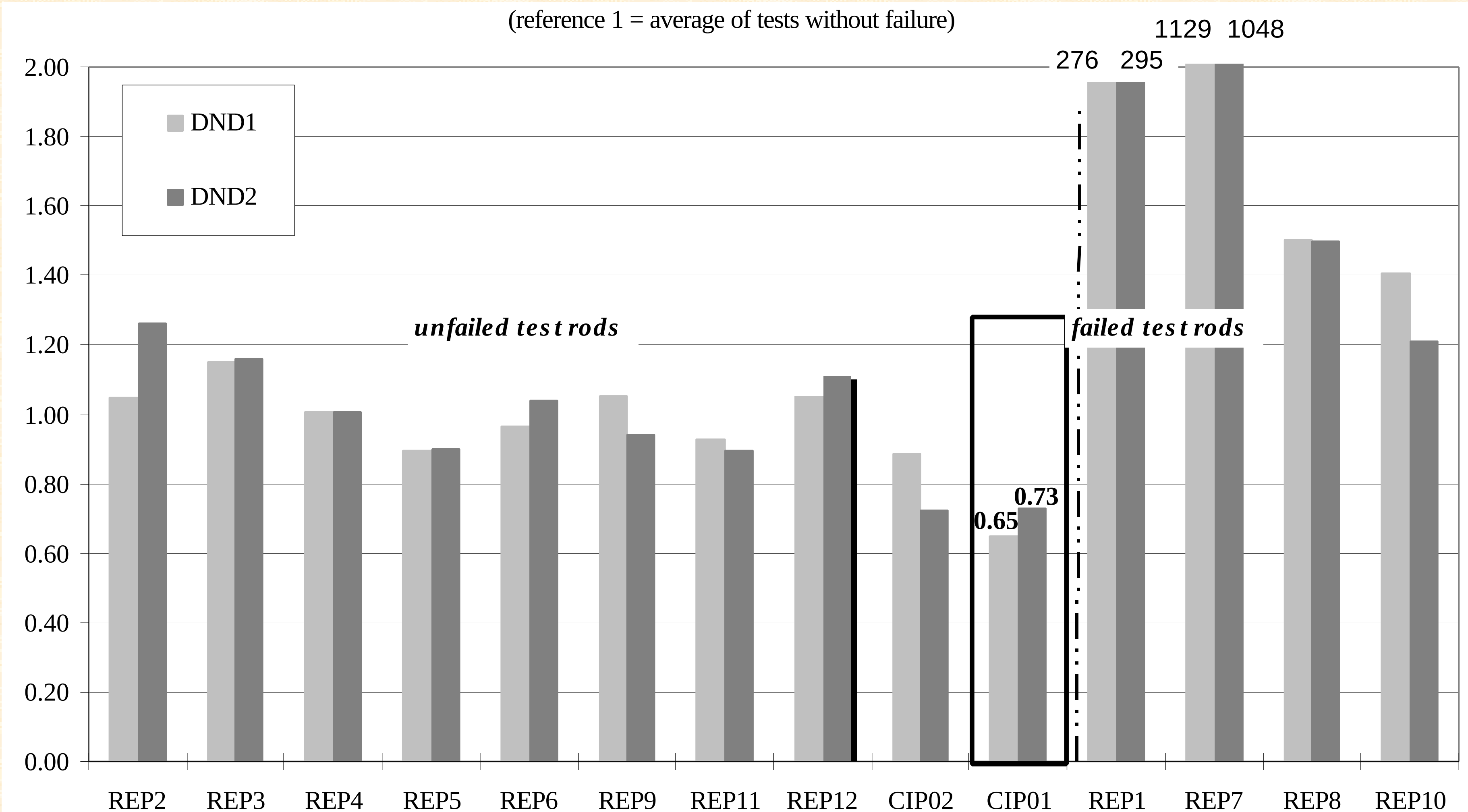
# CIP0-1 late event (4)

AEN NEA

- Microphone event close to saturation on M2
- Flow, pressure and VD event
- This event could be interpreted as clad failure, but:
  - No detection on DND signals
  - No detection of $^{85}$Kr on the Na cover gas after test
  - Unconsistent timing: P,Q before microphones
  - No failure seen during visual examination in hot cells
- Quantitative gamma-scanning on the upper plenum was performed

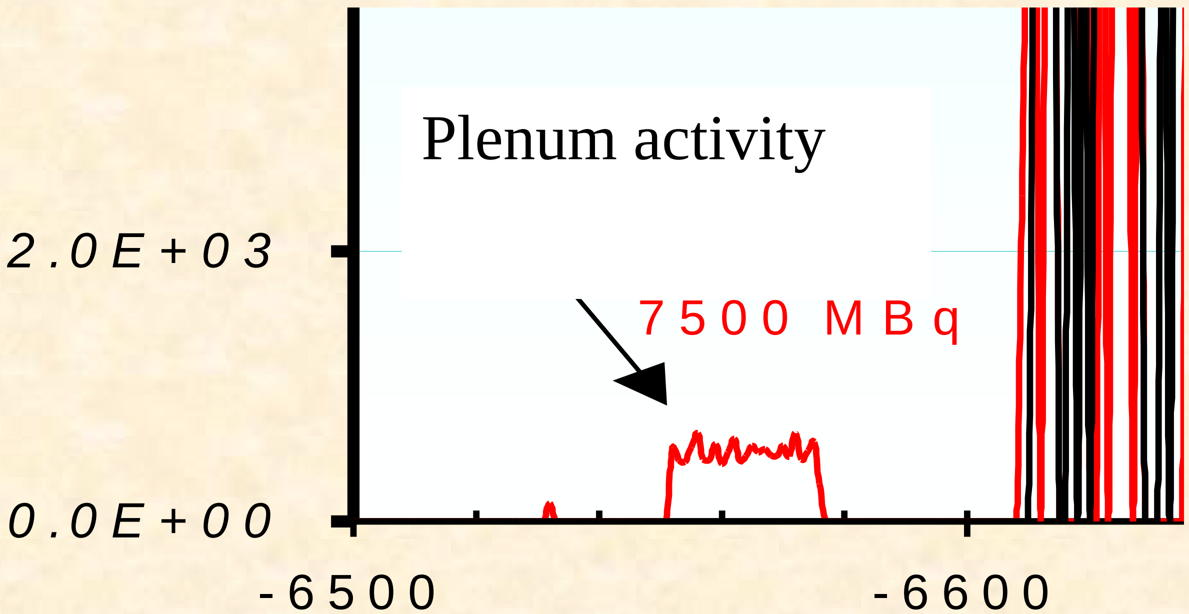
CIP

# REPNa and CIP0 DND



# CIP0-1 post-test NDE

Quantitative Gamma-scanning




**Activity consistent with expected amount of FG release**

CIP

# CIP0-1 results



- First event = clad elongation
- Second event = most probably not a failure

**The non failure will be confirmed by pin piercing in fall 2003**

Non-destructive examinations performed :

- visual examination, profilometry
- gamma-scanning
- zirconia layer : **extended spalling**

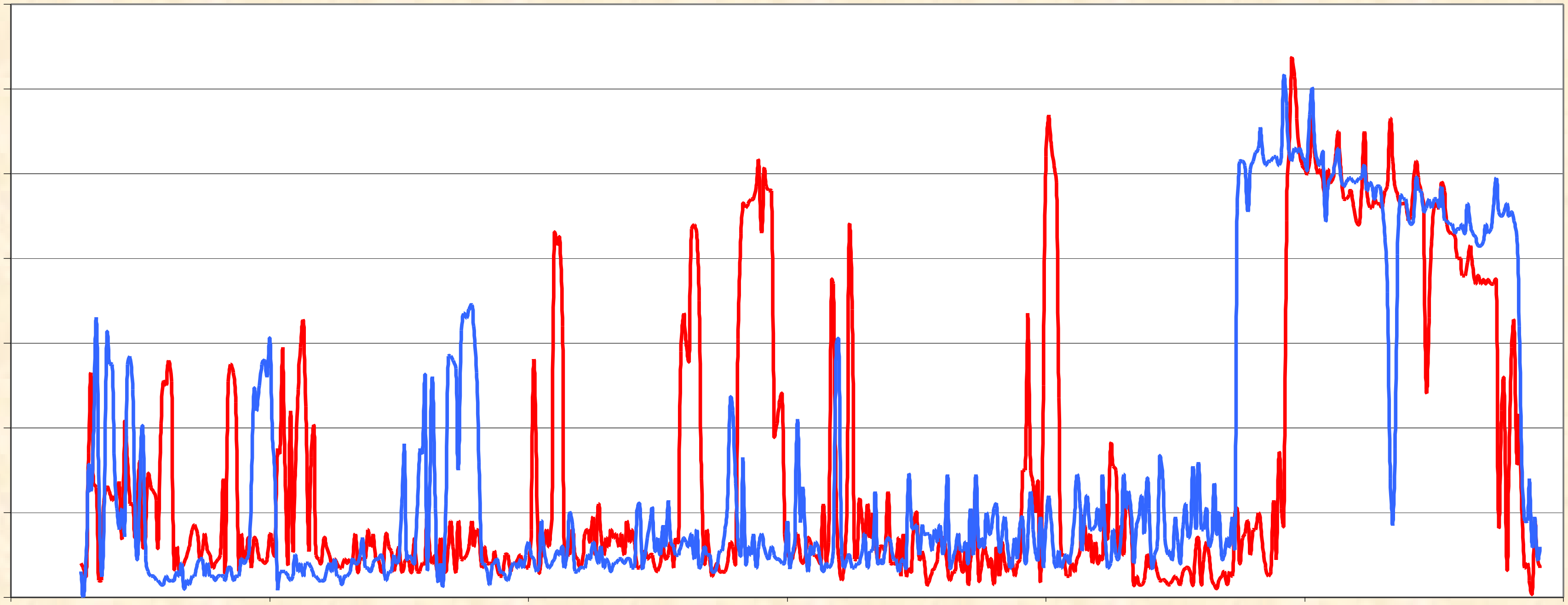
CIP

# CIP0-1 post-test NDE



## Zirconia measurement

**extended spalling**



# CIP0-1 status

- Pin piercing and gas analysis in **Fall 2003**
- Destructive examinations will be performed :
  axial and radial cuts at the **beginning of 2004**
- The signal analysis is undergoing (explanation of second event)

**PROMETRA Program** defined within the CIP for advanced cladding materials (Zirlo, M5-6cycles)

**Objectives :** determine the stress-strain laws and failure data

**Common test matrix**

- 10 hoop tensile tests (doubled) : T= 280-800°C, 1 $s^{-1}$

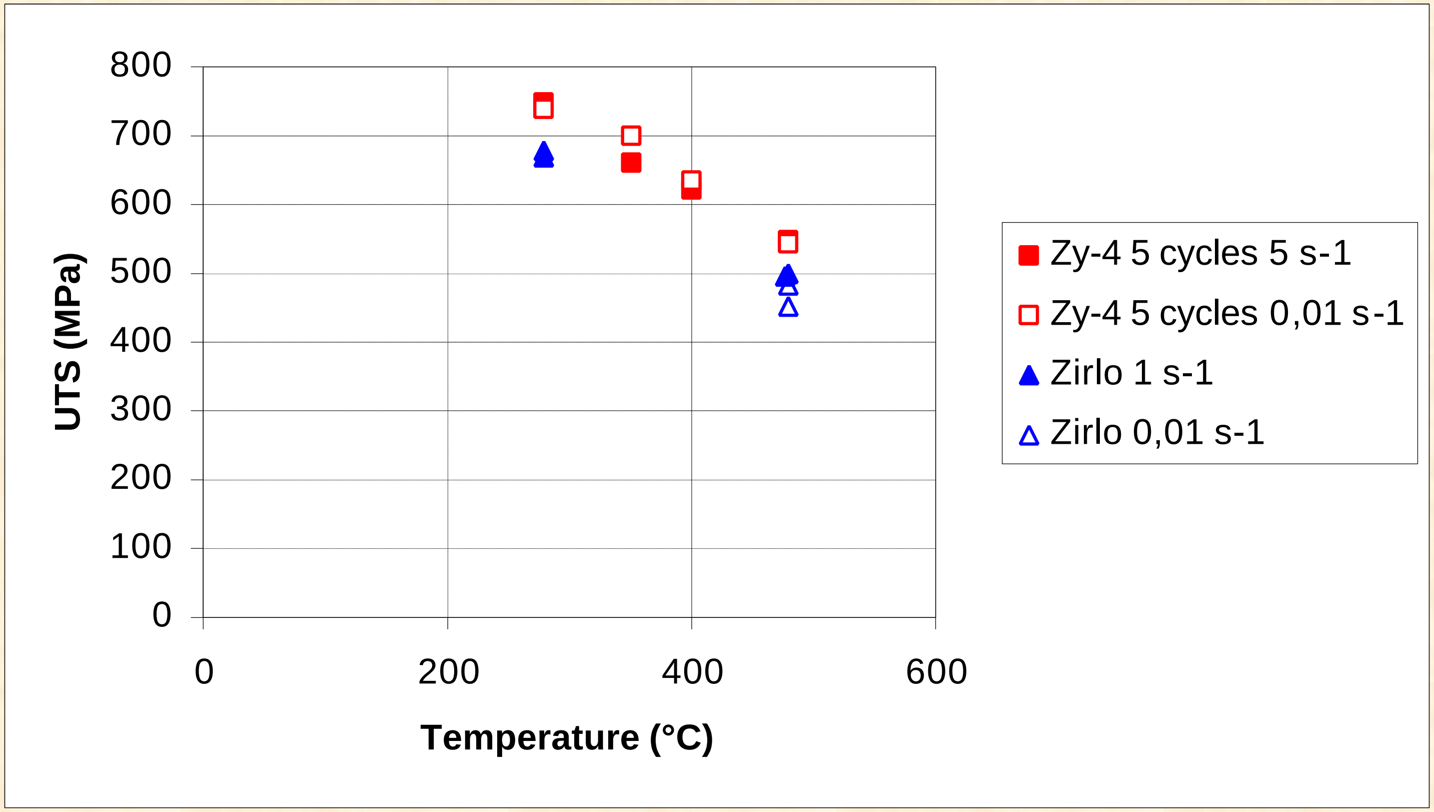
T=480°C, strain rate : 0.01 $s^{-1}$

- 8 Penn-State type tests (doubled) : T= 280-800 °C, 1 $s^{-1}$
- 2 burst tests T= 280°C, 1 $s^{-1}$

CIP

# MECHANICAL CHARACTERISATION OF ZIRLO CLADDING

## Comparison of UTS results between Zirlo and Zr4-5 cycles




Hoop tests at higher temperature finalised
Following part of the test program in 2004

# CONCLUSIONS

- CIP0-1 successfully performed
- Presumption of non failure – confirmation Fall 2003
- Physical origine of late event signals to be analyzed
- CIP0-1 very last test with Na loop
- 1st test (CIPQ) in the water loop foreseen in 2006

# Grid Reliability Issues

NRC Office of Nuclear Regulatory Research

William S. Raughley,
Sr. Electrical Engineer

# Agenda

- Background
- Changes on the Grid
- Impact on Nuclear Facilities
- Best Practices
- Conclusions

# Background

- Long term review of operating experience
- Monitoring deregulation
- SECY-99-129, "Effects of Electric Power Industry Deregulation on Electric Grid Reliability and Reactor Safety," May, 1999.
- "Operating Experience Assessment-Effects of Grid Events on Nuclear Power Plant Performance," April, 2003.
    - Initial draft for internal comment 11/02
    - Issued for stakeholder comment 5/03
    - NUREG scheduled 11/03

# Background - Expectations

- 10 CFR 50 App. A (GDC-17, Electric Power Systems)
- Station Blackout Rule (10 CFR 50.63)
- Maintenance Rule (10 CFR 50.65)

# Background

- In 1992, the National Energy Policy Act (NEPA) encouraged competition in the electric power industry, i.e., open generator access to transmission system and statutory reforms to promote wholesale generators.

- In 1996, FERC issued orders requiring open access to the electric power transmission system.

- Currently, 50% of states sell power in an open market, and have restructured to promote separate generating, and transmission and distribution companies.

# What Has Changed on the Grid?

- **Higher transmission system loading.**
- **Lower grid reactive capabilities.**
    - PJM 1999 grid event
    - Reactor power uprates
- **Changes in grid operating voltage limits and action levels**.
- **Increase in transmission line relief requests during summer.**
- **Increase in coordination times to recover from grid disturbance.**

# Challenging Safety Issues

- Most loss-of-offsite-power (LOOP) events occur during summer months.
- Increase likelihood of reactor induced LOOPs during summer.
- Longer time to recover from LOOP.
- Risk from
  - Low voltage condition
  - On-line EDG maintenance

# Best Practices

- Consider the seasonal effects of grid performance on:
  - EDG maintenance and test practices
  - Switchyard maintenance practices
- Establish contractual arrangements between grid operators and NPPs to maintain secure electrical power.
- Use of real-time grid parameters when performing design basis electrical analysis.

# Stakeholder Comments

- Recognized the merit in periodically conducting grid assessments.

- NRC and industry should collaborate on an assessment of the relationship between deregulation, grid events, and NPP safety.

- Grid operators may not be fully aware of:

  - the more restrictive NPP bus voltage limits,
  - the grid condition during EDG maintenance,
  - pre-trip voltages necessary for safe shutdown.

# Conclusions

- **Changes in grid performance have occurred since operating in a deregulated environment.**

- **Grid performance can impact NPPs:**

  - Response to accidents and transients
  - Blackout (coping) duration
  - Challenge safety equipment

- **Need to seek a better understanding of grid performance.**

# Overview of Research Findings on the Mitigating Systems Performance Index (MSPI)

## 2003 Nuclear Safety Research Conference
## October 20-22, Washington, DC

by

D. A. Dube

Operating Experience Risk Analysis Branch

Division of Risk Analysis and Applications

Office of Nuclear Regulatory Research

U.S. Nuclear Regulatory Commission

# Topics

- Description of MSPI
- Pilot Program Objectives
- Results and Lessons Learned
- Status of Key Technical Issues

# Background

- **MSPI evolved from feasibility study of Risk-Based Performance Indicators (RBPI) in NUREG-1753**
- **MSPI addresses recognized issues with current PIs**
- **MSPI is highly risk informed simplification to RBPIs with the following features:**
  - **Unavailability and unreliability consistent with PRA**
  - **Accounts for plant specific design and performance data**
  - **Eliminates fault exposure time**
  - **No cascade failure of cooling water support systems**
  - **Scope consistent with at-power internal events level-1**
  - **Performance thresholds consistent with basis for current PIs.**

# Fundamental Expression for the MSPI

***MSPI = UAI + URI***

***where UAI = Unavailability Index***
***URI = Unreliability Index***

# Calculating the Change in Unreliability

$$URI = \frac{\text{delta CDF}}{\text{delta unreliability}} \quad * \quad \text{change in unreliability from a baseline}$$

$$URI = B(UR) * \,)\, \mathrm{UR}$$

where *B* is the Birnbaum importance

# Linearization of Unreliability

$$URI = CDF_P \sum_{j=1}^{m} \left[ \frac{FV_{URcj}}{UR_{pcj}} \right]_{\max} (UR_{Bcj} - UR_{BLcj})$$

**where the summation is over those active components in the system that by themselves fail a "train"**

$CDF_p$ **is the plant-specific internal events, at-power core damage frequency,**

$FV_{URc}$ **is the component-specific Fussell-Vesely value for unreliability,**

$UR_{pc}$ **is the plant-specific PRA value of component unreliability,**

$UR_{Bc}$ **is the current estimate of ("Bayesian corrected") component unreliability for the previous 12 quarters,**

$UR_{BLc}$ **is the historical baseline unreliability for the component.**

# MSPI Pilot Objectives

- **Exercise MSPI Guidance:**
  - System boundary and component identification
  - Data collection
  - MSPI computation

- **Validation and Verification:**
  - Issue identification & special studies
  - SPAR model comparisons
  - Duplication of Pilot Plant results
  - Comparison to SDP (Table top)

- **Perform Temporary Instruction Inspections**

# List of MSPI Monitored Systems

| BWRs | PWRs |
|---|---|
| HPCI/HPCS (high pressure coolant injection/core spray) | HPSI (high pressure safety injection) |
| RCIC (reactor core isolation cooling) | AFW (auxiliary feedwater or equivalent) |
| RHR (residual heat removal) | RHR |
| EAC (emergency AC power) | EAC |
| Support System Cooling (ESW + CCW) | Support System Cooling |

# Plants Participating in MSPI Pilot Program

**Region I**
Limerick 1&2
Millstone 2&3
Hope Creek
Salem 1&2

**Region II**
Surry 1&2

**Region III**
Braidwood 1&2
Prairie Island 1&2

**Region IV**
Palo Verde 1,2&3
San Onofre 2&3
South Texas 1&2

# MSPI Calculation and Performance Thresholds

- Based on 12 quarters rolling averages of train unavailabilities and reliability data for six systems
- Thresholds of performance consistent with NRC policy and set at
  - GREEN for change in MSPI less than 1E-6
  - WHITE for change of 1E-6 to 1E-5
  - YELLOW for change of 1E-5 to 1E-4
  - RED for change of greater than 1E-4

# Overall MSPI Results

- SPAR Rev 3 models were not adequate to verify risk model importances at the component level. A major SPAR enhancement effort was undertaken.
- Differences between the Plant PRA and SPAR Resolution Models are explainable. However, this MSPI verification task should not be construed to validate the Plant PRA.
- Overall, MSPI results using Pilot Plant models and SPAR resolution models are in very good agreement.
  - Two identical valid WHITE conditions out of one hundred systems for $4^{th}$ Qtr 2003
  - Numerical results (values above >1E-7) generally agree within factor of three
- Because of the close agreement discussed above, it is believed that there were no major systematic deficiencies in the Pilot Plant MSPI submittals with the exception of an early spreadsheet error that was corrected.

# Lessons Learned from Pilot

- **Additional guidance is needed on calculation of baseline Unavailability (e.g. treatment of unusually long planned train outage in 3-year historical period)**
- **Additional guidance on modeling standby versus normally running trains is recommended**
- **Additional guidance is needed regarding components with no available FV that otherwise would be monitored**
- **Additional Quality Control on data entry is needed. Software should have internal checks to reduce errors.**

# Key Technical Issues

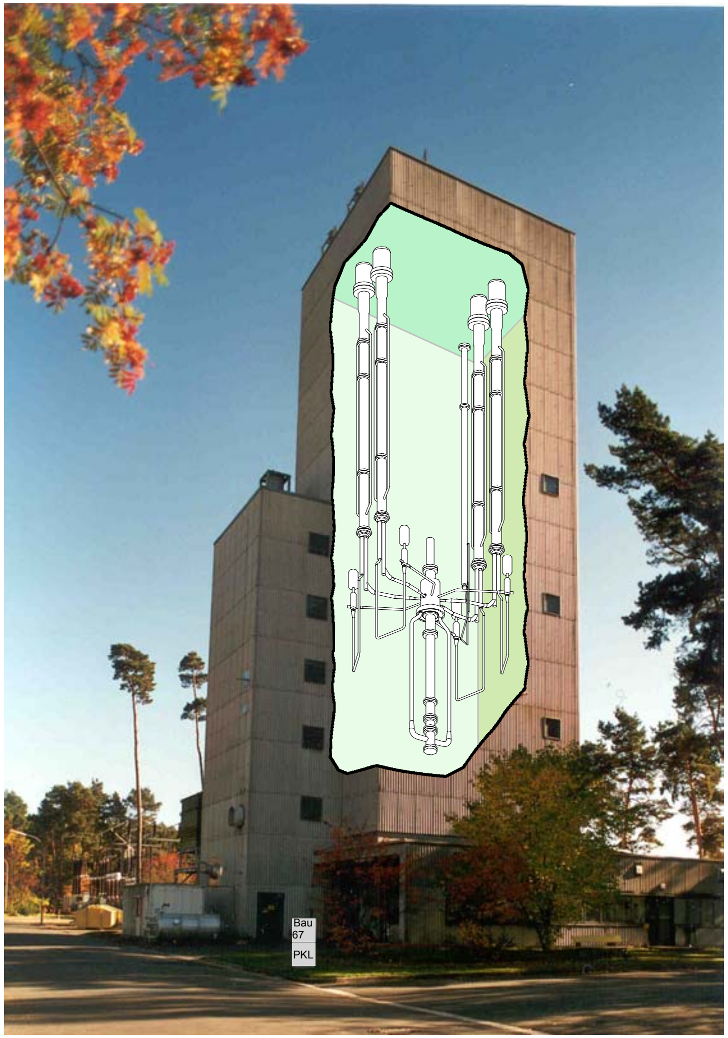
- **Verification – significant differences between SPAR model and Plant PRA at the MSPI component level.**
- **One component failure resulting in system indication turning to WHITE (“Invalid Indicator”).**
- **Large number of failures to turn system to WHITE (“Insensitive Indicator”).**
- **Identification of system boundaries.**
- **Data Issues including generic industry performance data, and data collection burden.**
- **Support system contribution to Fussell-Vesely.**
- **Treatment of Common Cause Failure contribution to Fussell-Vesely.**

# Status of Pilot Program

- Issued Program Guidelines in NEI 99-02 Revision and Regulatory Issue Summary in September 2002.
- Licensees collected performance data Sept 2002 through Feb 2003 and submitted to NRC on monthly basis.
- NRC Issued Temporary Instruction and performed inspections of Pilot Plant implementation.
- Completed a major effort to reconcile differences in SPAR and Plant PRAs for 11 distinct models (all 20 units in Pilot).
- Preliminary research report on Pilot Program drafted for internal NRC review, and Public Review and Comment late 2003 through mid 2004.
- Address implementation issues Fall 2003.

# Summary

- MSPI is a risk-informed performance indicator using plant-specific design configuration and equipment performance data.
- Resolutions to the key technical issues have been proposed.
- Minimum and maximum limits to the number of component failures before the component is deemed degraded (non-GREEN) will be effected (frontstop and backstop concepts).
- Verification effort identified that SPAR Rev. 3 models are <u>not</u> adequate for verifying risk measures at the component level. SPAR models were improved.
- Additional implementation issues will be addressed through the late ’03 and early ’04.




# Boron Dilution Tests / PKL

Klaus Umminger, Bernhard Brand
Framatome ANP, Germany

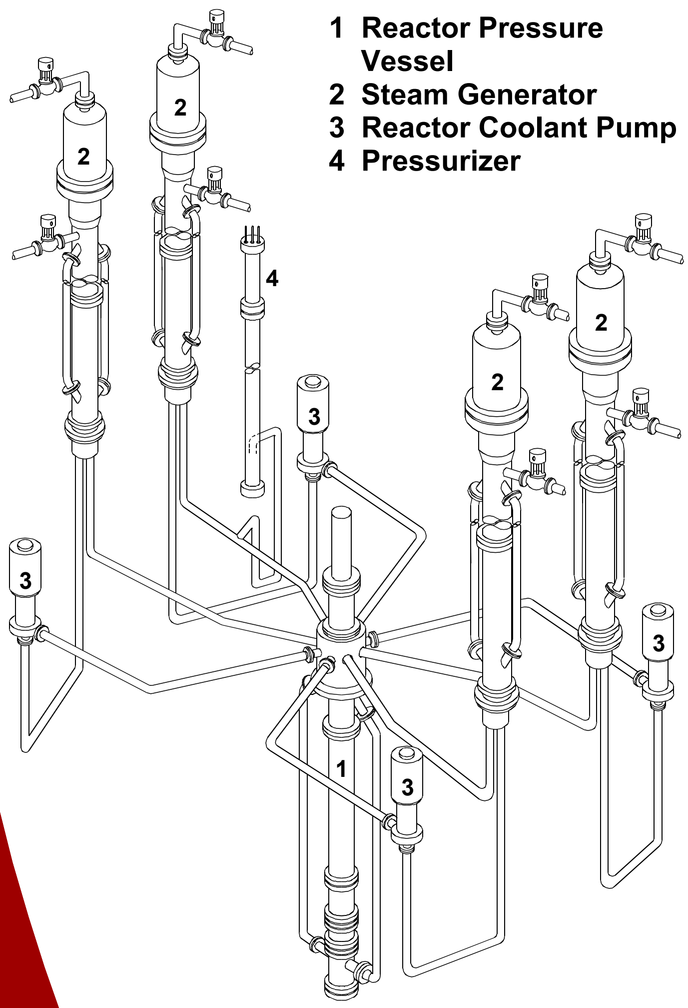
Nuclear Safety Research Conference
October 20-22, 2003
Washington DC, USA

# Boron Dilution Tests / PKL

## Outline

- PKL Test Facility
- Current PKL / OECD Project
- PKL Test Results on
  - Boron Dilution after Small Break Loss of Coolant Accidents
  - Loss of Residual Heat Removal under shut-down Conditions
- Conclusions and Outlook

# PKL III Test Facility




## Integral test facility simulating a 1300 MW PWR

- 4-Loop configuration
- All relevant safety and operational systems on the primary and secondary side without turbines and condensers
- Volume and power scale 1:145 (Diameter 1:12)
- Elevations 1:1

## Objectives

- Thermal hydraulic system behavior under accident situations
- Verification and Optimization of cooldown procedures
- Database for the validation of computer codes
- Demonstration of safety margins
- Training of the operating staff
- Test Facility available at short notice for topics of current intere



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# Overview on the current PKL / OECD Project

- **The current PKL program is included in the international progran SETH, which was set up by the OECD/Nuclear Energy Agency**

- **Partners in the PKL/SETH program**
  - Germany (German utilities, GRS, Framatome ANP): 50 % contribution
  - 14 OECD member countries (e.g. USA, Japan, France): 50 % contribution

- **Topics of investigations within the PKL/SETH program**
  - Boron dilution after small break loss of coolant accidents (SB-LOCAs)
  - Loss of residual heat removal system (RHRS) under shut-down conditions

  All experiments have been performed with original boric acid and adequat measurements for the detection of the boron concentration

- **The PKL/SETH program will be completed at the end of 2003**

# PKL-Tests on Boron Dilution

**Background:**

- Low borated water entering the core can lead to local recriticality and thereby to a power excursion

- A necessary condition for this to happen is the formation of low borated water slugs in the primary and their transport to the core without sufficie mixing

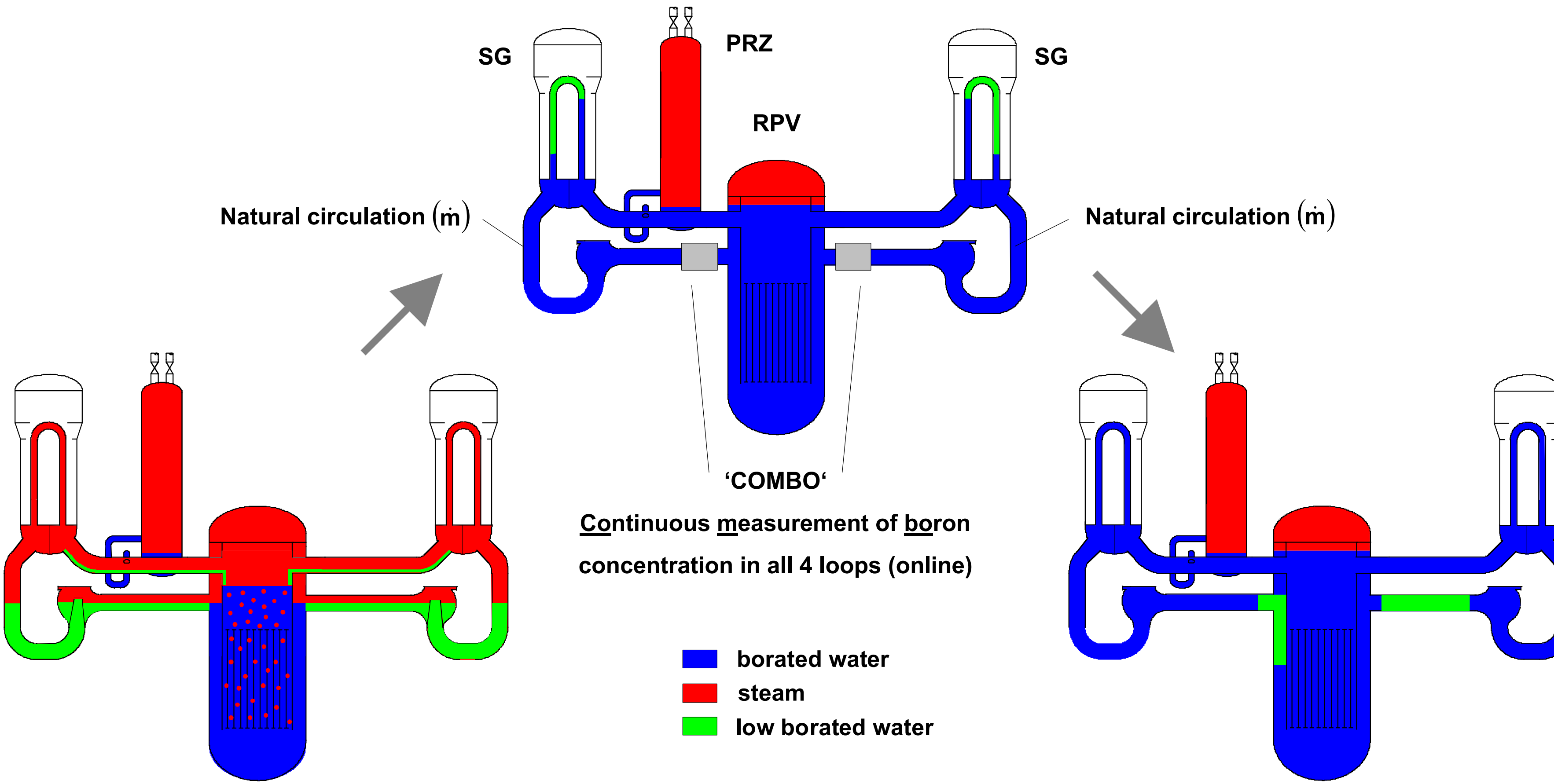
- Formation of low borated water slugs might occur by separation of borat and almost boron free coolant within the primary system, e.g. by

→ **SB-LOCA and Reflux-Condenser Conditions**

→ **Loss of RHRS under shut-down conditions** followed by steam production in the core and condensation in the SG

# Transport of Low Borated Slugs of Water after SB-LOCA

## *Schematic illustration, without taking credit from mixing*




a) During reflux-condenser phase

b) After refill of the RCS

c) After restart of natural circulatio

# PKL Tests on Inherent Boron Dilution after SB-LOCA

## General Topics of Investigation

- **Formation of slugs of low boron concentration in the RCS during reflux-condenser mode after SB-LOCA**
  - size of low borated water slugs
  - boron concentration within the slugs
- **Mixing of differently borated water flows during primary side fillup and after startup of natural circulation**
  - within the reactor coolant lines
  - within the steam generators
- **Restart of natural circulation after refilling of the primary system**

➢ **Transport of low borated water into the reactor pressure vessel**

**Amount, boron concentration, velocity, differences between the loops**

# PKL Tests on Inherent Boron Dilution after SB-LOCA

## Relevant Parameters

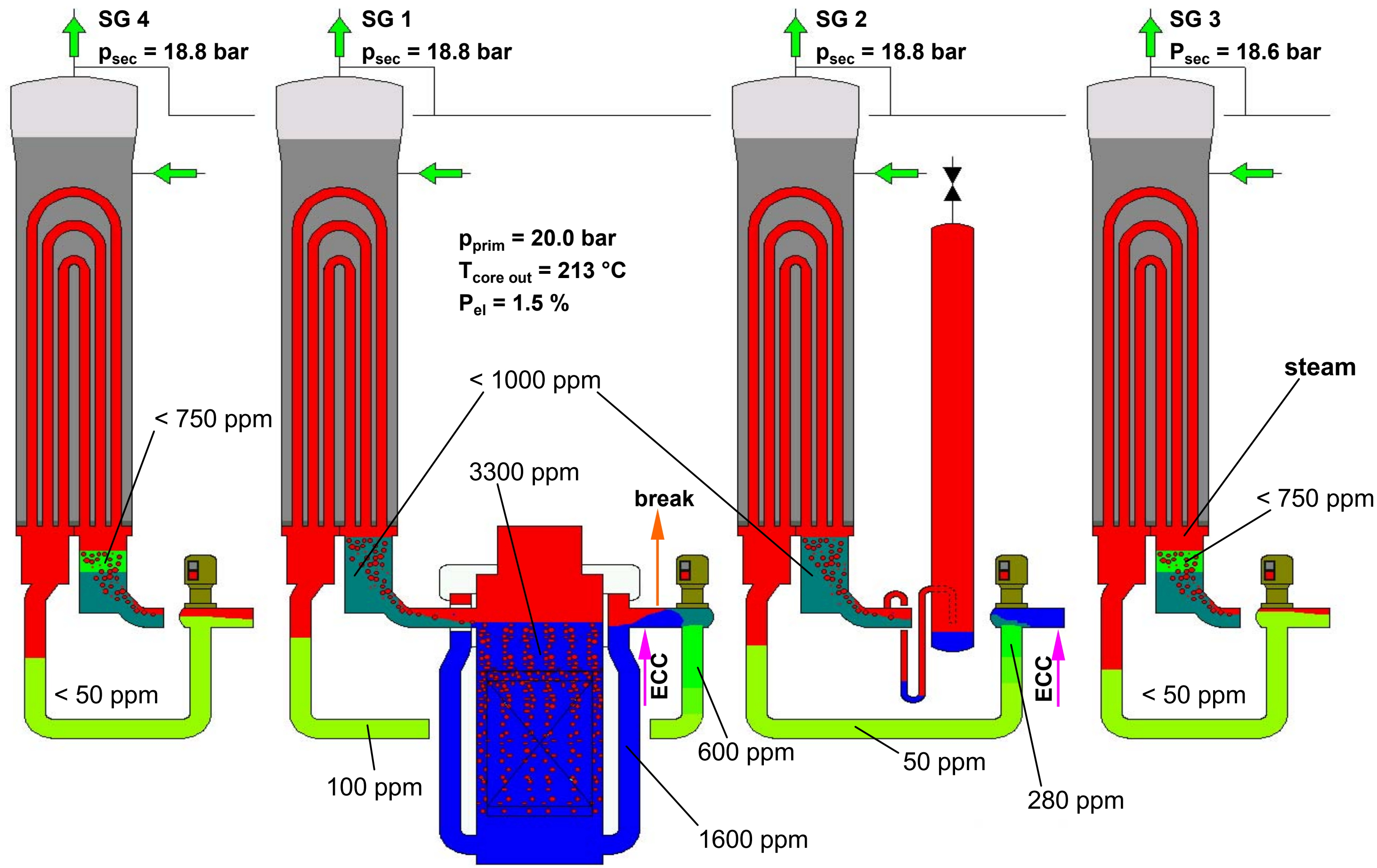
- Break size and break location
- Injection mode (number of available systems, location, injection rates)
- Other boundary conditions such as cool down rate, extra borating system

## Conservative assumptions:

**Boundary conditions, so that restart of natural circulation (NC) results in the boron concentration at the RPV inlet (and in the core) to decrease extremely**

- Either by favoring simultaneous restart of NC in all 4 loops (finally simultaneou arrival of the condensate slugs at the core) by high symmetry between the loo
- Or by favoring the formation of extremely large slugs  of condensate before restart of NC (long reflux condenser phase and high levels due to a certain break size / ECC-injection configuration), e. g. test E2.2

# PKL III E2.2: Distribution of Water and Boron at Minimum Primary Coolant Inventory




- Break: Cold leg (32 cm², Loop 1)
- ECC-injection: 2 of 4 SIPs into the cold legs (Loop 1 and 2, 2200 ppm)
- 100 K/h cool-down of the steam generator secondaries
- Assumption: Homogeneous boron concentration in the RCS at occurrence of the break - 1000 ppm



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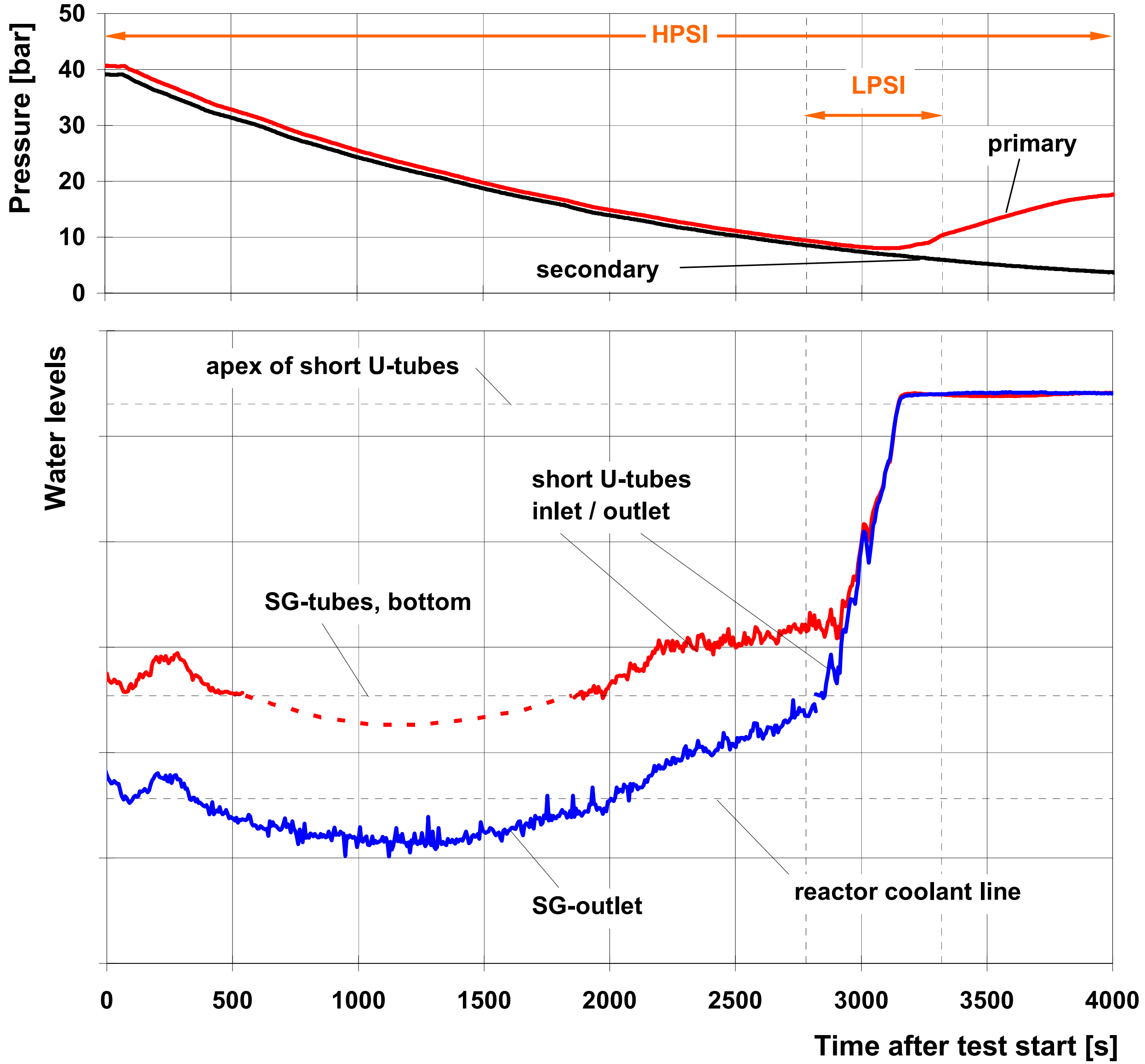
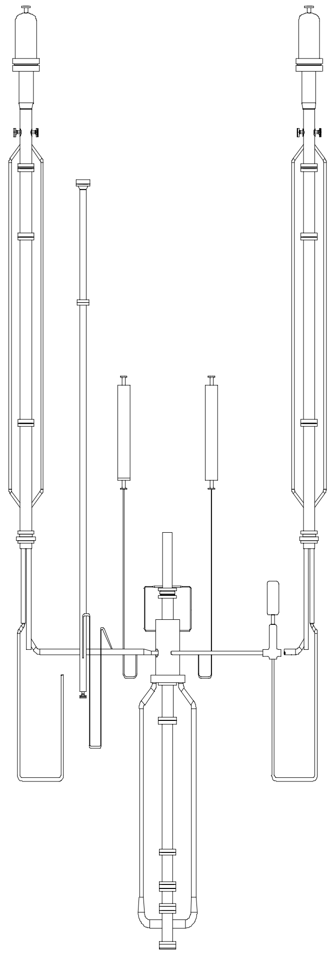
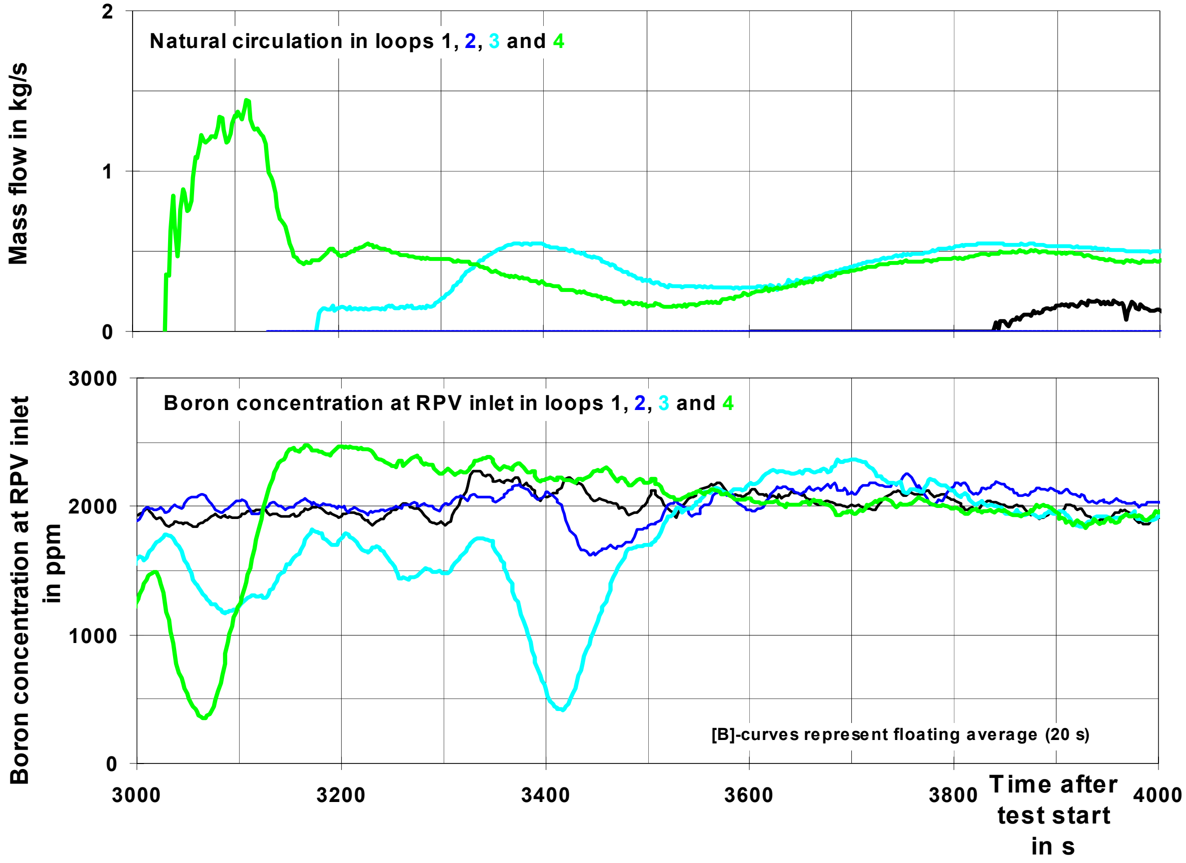
# PKL III E2.2: Cooldown and Refill



# PKL III E2.2: Transport of Condensate Slugs due to Restart of Natural Circulation




## Test results:

- Restart of natural circulati after refill occurs at differe times and different intensities for different loo
- Slugs of condensate from different loops reach the RPV at different times
- Effective mixing of condensate and borated water in the loops with EC injection

# PKL Tests Concerning Boron Dilution after SB-LOCA

## Conclusive Findings up to now

- Formation of low borated water slugs under reflux condenser conditions following SB-LOCAs experimentally confirmed
- Maximum size of low borated water slugs smaller than expected.
- Significant but short term reduction in boron concentration in case of ECC-injection into 2 of 4 loops (cold leg) in the loops which are not supplied with ECC-water
- Effective mixing in the loops and the SGs during the refilling process and during the transport of slugs towards the RPV (minimum boron concentration at the RPV inlet: 350 ppm)

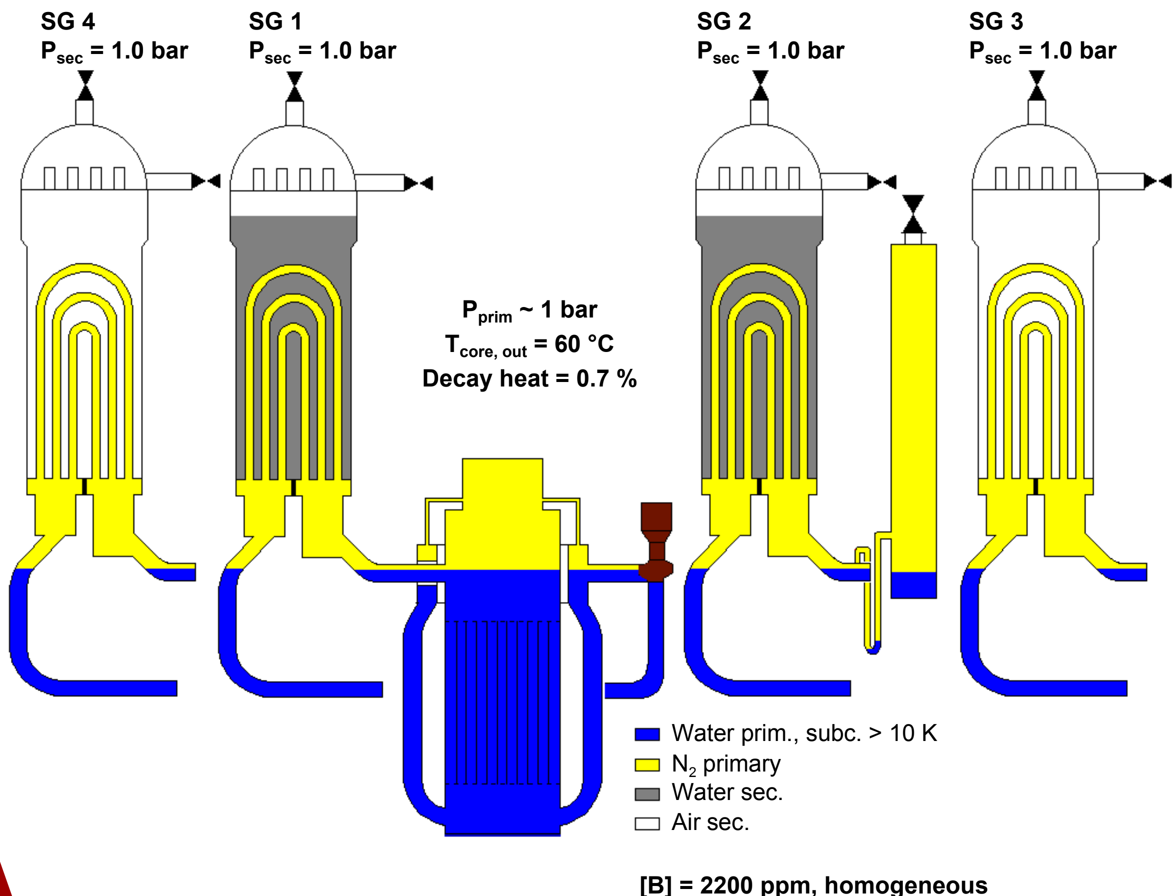
# PKL Tests Concerning Boron Dilution after SB-LOCA

## Further Needs for Experimental Investigations

- Investigations concerning the influence of specific boundary conditions and plant configurations (e. g. ECC injection location and rates, cool-down rates) on
  - Natural circulation behavior
  - Maximum boron concentration at the RPV inlet
- Systematic investigations on the transfer of borated water from the SG inlet to the SG outlet side preventing the accumulation of low borated water slugs
  - Influence of the water inventory in the steam generator tubes
  - Influence of the primary pressure and the cool-down rate

# PKL III E3.1: Loss of RHRS

## Initial Conditions




## Scenario

- Total failure of RHRS when the reactor is shut down, e.g. for refueling (primary system still closed)
- 2 SGs filled with water, one of both (SG 1) ready for operation

## Objectives

- Takeover of residual heat removal by the SG ready for operation
- Evolution of local boron distribution



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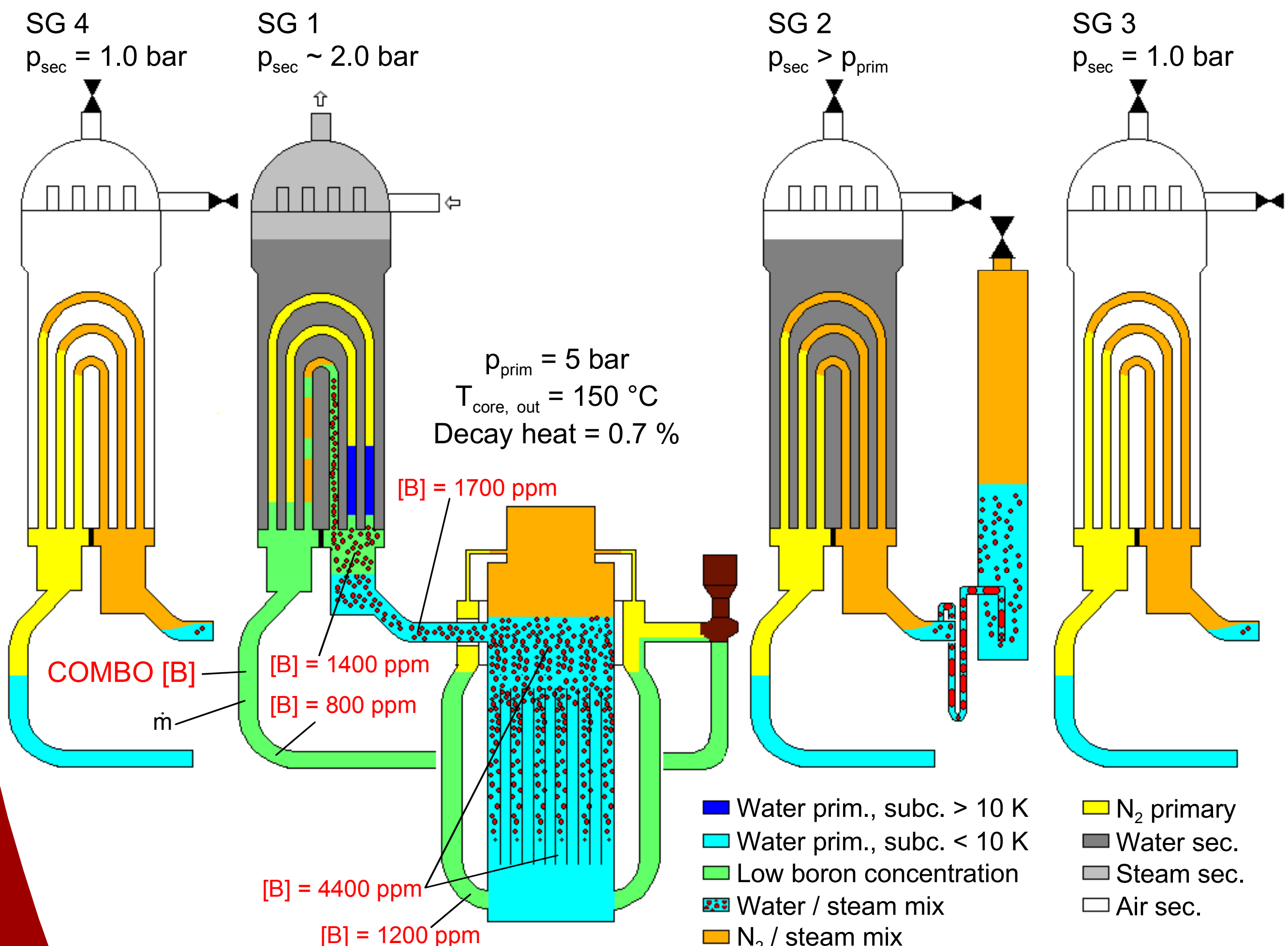
# *PKL III E3.1: Loss of RHRS*

**Situation after 7 - 8 hours**



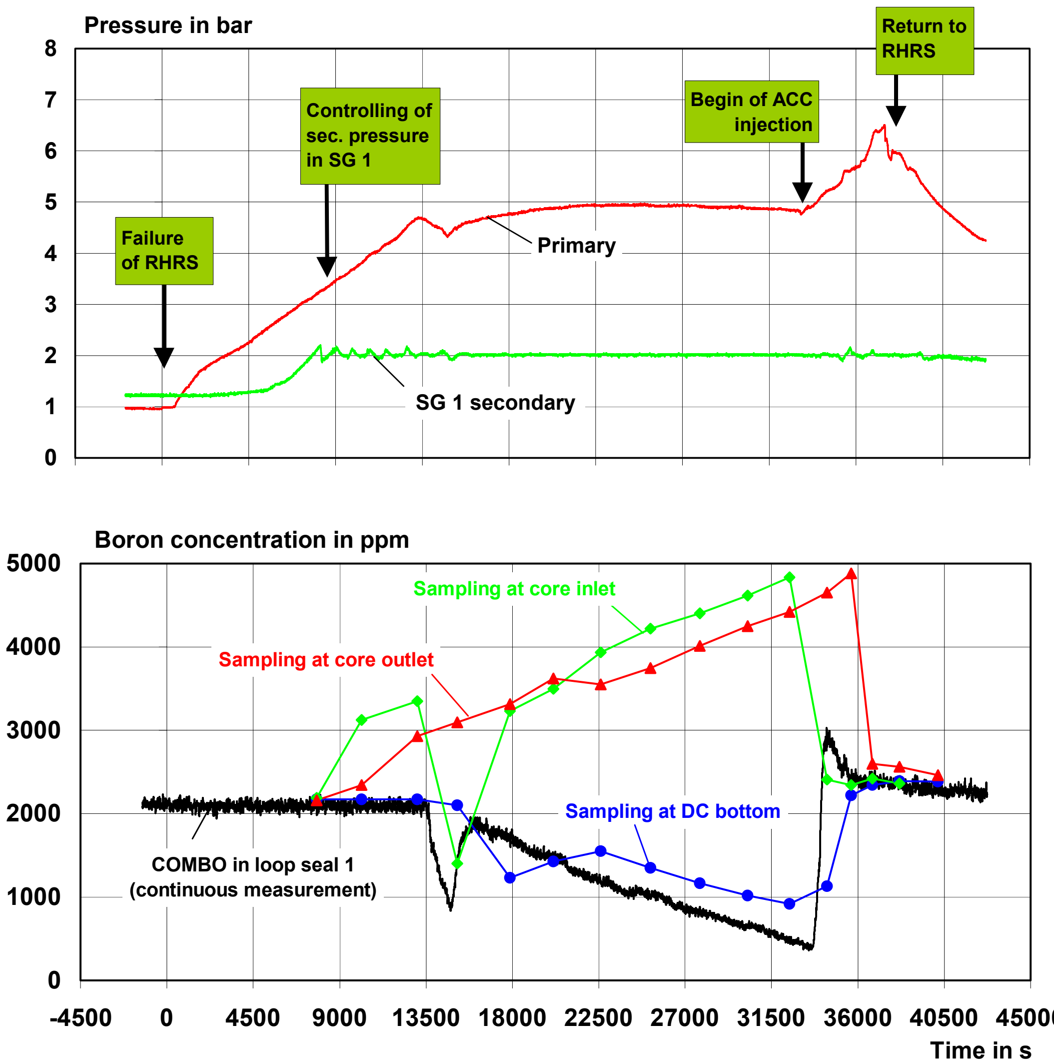

**Failure of RHRS has lead t**

- **Core heat up and steam production**
- **Increase in liquid level and primary pressure**
- **Condensation in the SG**

**Heterogeneous behavior ir the individual SG U-tubes:**

- **Slug of subcooled wate in the long U-tubes**
- **Intermittent overflow of low borated condens in the small U-tubes**

# PKL III E3.1: Loss of RHRS




- Primary pressure stabilizes after about 5 h at 5 bar
- Heat transfer by steam production in the core and condensation in the SG in operation
- Intermittent overflow of condensate (low borated water) to the cold side in some SG U-tubes

➔ **Slow but continuous decrease in boron concentration at the SG outlet and with some delay also in the RPV downcomer**



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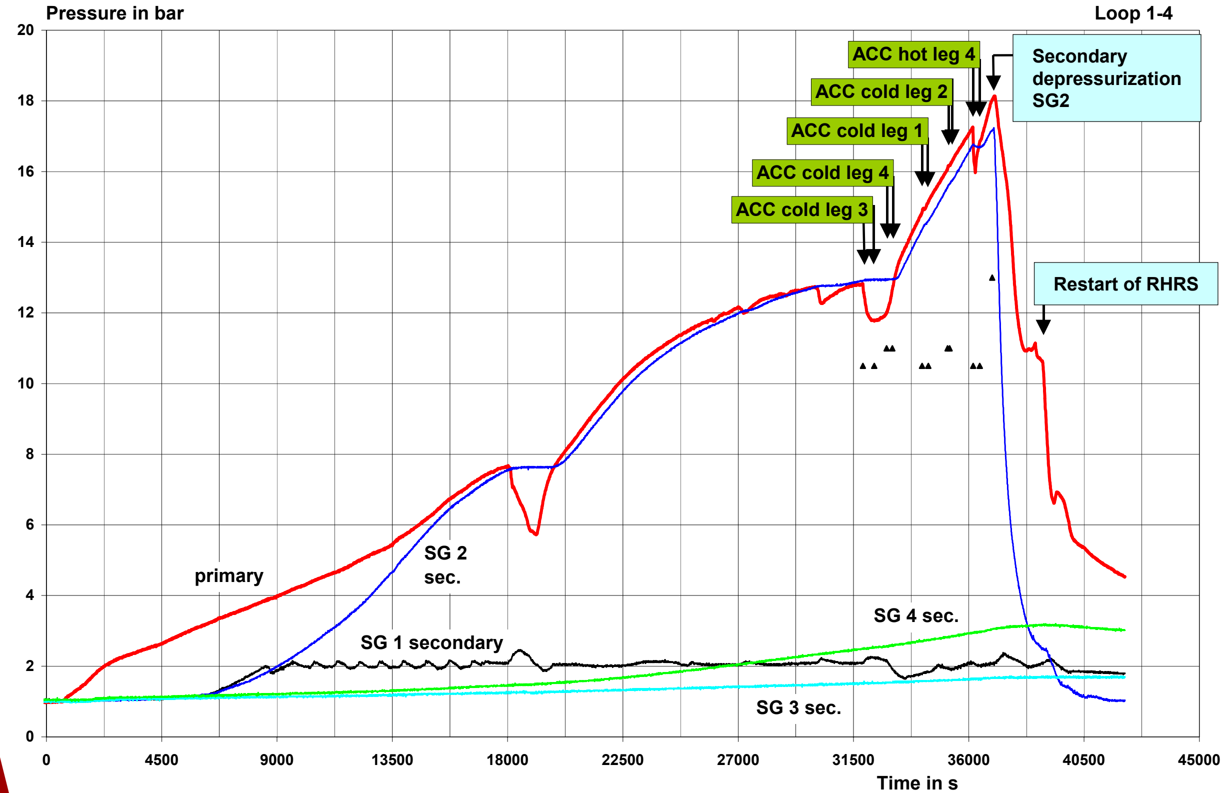
# *PKL III E3.1-IBS2 – Loss of RHRS (Hot Pressurizer at Test Start)*




**PKL III E3.1 - IBS2: Pressures on primary and secondaries**

## Scenario

**Higher temperature in the pressurizer at test start, otherwise identical conditions as in test E3.1**

## Results (compared to E3

- **Smaller amount of water displaced into the pressurizer**
- **No or less SG-tubes are "Blown Through"**
- **Increase in primary pressure over 10 bar**
- **Smaller amount of low borate water displaced from the SG-inlet to the SG-outlet**



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# PKL Test Concerning Loss of RHRS

## Conclusive Findings up to now

- Comparison of test E 3.1 (cold pressurizer) with commissioning test E 3.1-IBS 2 (hot pressurizer) shows significant differences with respect to the evolution of the primary pressure and boron dilution phenomena

  - Primary pressure stabilization at about 5 bar, however significant reduction of the boron concentration on the 'cold side' in test E3.1
  - No significant reduction of the boron concentration, however increase of the primary side pressure over 10 bar in commissioning test E3.1-IBS2

## Further Needs for Experimental Investigations

- Systematic investigations concerning the influence of the primary side inventory (e.g. influenced by state of the pressurizer) and the number of available steam generators on primary pressure and boron concentration

# *Conclusions and Outlook*

- **The experiments performed within the PKL / SETH program provide a unique database for boron dilution events and accidents under shut down conditions**
  - Important contribution to the resolution of safety issues of PWRs
  - Valuable database for the validation of thermal hydraulic system codes

- **Further needs for investigations have been identified and are planned to be addressed within a new OECD program on PKL**

- **The proposed investigations, which have been agreed with the SETH partners, also include new topics such as loss of residual heat removal with <u>open</u> RCS and boron dilution after steam generator tube rupture**

# Safety Culture Inspection at Davis-Besse

NSRC

October 21, 2003

J. J. Persensky

DSARE/RES

# Operating Experience and Davis-Besse

- NRC, DBNPS, and the industry failed to review assess, and follow-up on relevant operating experience
- DBNPS failed to assure that safety issues would receive appropriate attention
- NRC failed to integrate known information into its assessments of DBNPS

# Operating Experience and Davis-Besse

- 30 years of experience involving material wastage of components due to primary system leaks
- 17 NRC Generic Communications on BAC between 1980 and 2002
- 18 documents on BAC issued by INPO between 1981 and 2002

# Operating Experience and Davis-Besse

- Mind set developed that BAC of the RPV head would not result in wastage because of high temperatures
- Past lesson - inability to predict environmental conditions, especially inside containment

# Operating Experience and Davis-Besse

- DBNPS failed to address the lessons learned from a precursor event in 1998
- DBNPS did not include LERs or trending of LERs within scope of Operating Experience program
- DBNPS personnel were unaware of NUREG/CR-6245
- DBNPS personnel were generally unaware of operating experience at other PWRs
- DBPNS did not adequately review, assess or effectively correct leakage events
- Processing of external operating experience was not thorough or timely

# What does this all have to do with Safety Culture???

# Policy Statement on Conduct of Operations

- "Management has the duty and obligation to foster the development of a "safety culture" at each facility and to provide a professional working environment, in the control room and throughout the facility, that assure safe operations"

- "Management should review their procedures and policies on the conduct of operations …. to assure they support an environment for professional conduct."

# Safety Culture Definition

- " . . . That assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance."

- Policy Statement on Conduct of Operations (54FR3424) -1989
- INSAG-4

# INSAG 15 Attributes of Safety Culture

- Commitment
- Use of Procedures
- Conservative Decision Making
- Reporting Culture
- Challenging Unsafe Acts
- Learning Organization
- Underpinnings
  - Communications
  - Clear Priorities
  - Organization

# Learning Organization - Characteristics

- Operational experience – Internal & External
- Benchmarking - search for improvements and new ideas
- Monitoring and providing feedback
- Active involvement and teamwork
- Self-assessment

# DBNPS's Root Cause Analysis

- Staff and Management Exhibited Less than Adequate Nuclear Safety Focus
- Weaknesses Existed in Nuclear Safety Culture, Standards, and Decision-Making
- Management ineffectively implemented processes, and thus failed to detect and address plant problems as opportunities arose.

# Regulatory Basis

- 10 CFR 50, Appendix B, Criterion XVI, Corrective Action" requires that:
  - Significant conditions adverse to quality are promptly identified and corrected.
  - The cause of significant conditions adverse to quality is identified and actions are taken to preclude repetition.

- "The reactor head degradation was a significant condition adverse to quality requiring correction and action to preclude repetition."

# Inspection Plan

- Evaluate Licensee’s Internal and External Review Processes to assess safety culture
- Evaluate Licensee’s Long-Term Approach for monitoring continued safety culture improvement
- Evaluate Licensee’s Employee Concerns Program Effectiveness
- Evaluate Licensee’s SCWE and SCWE Review Team (SCWERT) Effectiveness

# Inspection Guidance

## NRC Guidance

- 1989 Policy Statement on Conduct of Nuclear Power Plant Operation
- 1996 Policy Statement on Freedom of Employees to Raise Safety Concerns Without Fear of Retaliation
- NRC Inspection Manual Chapter on Problem Identification and Resolution
- NRC Inspection Manual Chapter on Resolution of Employee Concerns

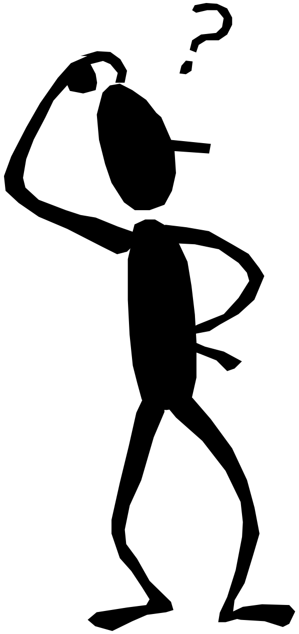
# Inspection Guidance

**Internationally Recognized Guidance/International Nuclear Safety Advisory Group (INSAG) Documents**

- INSAG 4, "Safety Culture"
- INSAG 11, "Developing Safety Culture in Nuclear Activities: Practical Suggestions to Assist Progress
- INSAG 13, "Management of Operational Safety in Nuclear Power Plants"
- INSAG 15, "Key Factors in Strengthening Safety Culture"

# Conclusion

- Failure to review assess, and follow-up on relevant operating experience contributed to the event at DBNPS
- Operating experience is a primary element of a learning organization
- The existence of a learning organization is an important element of Safety Culture
- Safety Culture was a major root cause of the event



# Questions?